DOE/NE-0007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF	)	
	)	
PROPOSED RULEMAKING ON	)	
THE STORAGE AND DISPOSAL	)	**PR-50,51**
OF NUCLEAR WASTE	)	**(44 FR61372)**
	)	
	)	
(Waste Confidence Rulemaking)	)	

STATEMENT OF POSITION OF THE UNITED STATES DEPARTMENT OF ENERGY

15 April 1980

DOE/NE-0007
UC-70

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
PROPOSED RULEMAKING ON THE	)	PR-50, 51 (44FR61372)
STORAGE AND DISPOSAL OF	)	
NUCLEAR WASTE	)	
	)	
(Waste Confidence Rulemaking)	)	

STATEMENT OF POSITION OF THE UNITED STATES DEPARTMENT OF ENERGY

15 April 1980

CONTENTS

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

LIST OF TABLES

LIST OF TABLES (continued)

LIST OF TABLES (continued)

LIST OF FIGURES

LIST OF FIGURES (continued)

LIST OF FIGURES (continued)

LIST OF FIGURES (continued)

LIST OF FIGURES (continued)

I INTRODUCTION

I.A BACKGROUND AND SCOPE OF THE PROCEEDING

The Nuclear Regulatory Commission (the Commission or NRC) has defined the scope of this rulemaking as follows (1):

> The purpose of this proceeding is solely to assess generically the degree of assurance now available that radioactive waste can be safely disposed of, to determine when such disposal or off-site storage will be available, and to determine whether radioactive wastes can be safely stored on-site past the expiration of existing facility licenses until off-site disposal or storage is available.

This rulemaking was initiated by the Commission in response to the decision of the United States Court of Appeals in State of Minnesota v. NRC (2). It also is a continuation of previous proceedings conducted by the Commission concerning its confidence as to whether methods of safe disposal of high-level nuclear wastes will be available when they are needed (3).

On 23 May 1979, the United States Court of Appeals for the District of Columbia Circuit remanded two licensing actions to the Commission to consider whether an off-site storage solution for nuclear wastes will be available by the expiration dates of the operating licenses of the Vermont Yankee and Prairie Island nuclear power plants, to which the Commission had granted permits to increase on-site spent fuel storage facilities, and, if not, whether spent fuel can be stored on the sites past those dates and until an off-site solution is available. The court did not set aside or stay the challenged license amendments.

The Commission, on 25 October 1979, issued a Notice of Proposed Rulemaking which commenced this proceeding (1). Pursuant to that Notice, the United States Department of Energy (the Department or DOE), on 23 November 1979, filed a Notice of Intent to be a Full Participant in this proceeding. On 22 January 1980, a prehearing conference was held by the Presiding Officer appointed by the Commission to monitor the early stages of the proceeding and to assist the Commission in conducting the later portions.

This Statement of Position is filed pursuant to the Order issued by the Presiding Officer on 1 February 1980 (4). In that Order, the Presiding Officer sustained the Department's position that this proceeding should consider, as the representative case for handling high-level nuclear wastes, disposal and storage of spent nuclear fuel taken directly from commercial power reactors. This approach is consistent with the fact that the Commission previously suspended its further consideration of reprocessing of spent fuel from commercial reactors, In the Matter of Generic Environmental Statement of Mixed Oxide Fuel, Docket No. RM-50-5, 42 Fed. Reg. 65,334 (1977); following the decision of the President on 7 April 1977 to defer indefinitely all civilian reprocessing of spent fuel. In view of the fact that the Commission need find reasonable assurance only that spent fuel in some form can be safely stored and disposed of by any single method and the need to avoid injection of extraneous matter into this already complicated proceeding, it is appropriate to address the direct storage and disposal of spent fuel as the representative case.*

The Presiding Officer also has ordered that this proceeding is concerned solely with high-level waste, and that issues of low-level waste, uranium mill tailings and the safety of transportation of waste materials are not within its scope (4). Consequently, there is no necessity to explore here these other issues, which have been or are being considered by the Commission in other proceedings.

I.B DEPARTMENT OF ENERGY AUTHORITY AND EXPERIENCE

The Department of Energy has the statutory mandate and lead responsibility to conduct for the Federal Government research concerning nuclear

*Assessing confidence in the ability to store and dispose of spent fuel does not prejudge in any way using other approaches, such as the reprocessing of spent fuel and solidification of resultant nuclear wastes. In view of continued interest in the potential performance of specially processed waste forms in providing long-term isolation in disposal facilities, a summary of the status of research and development programs considering these alternative waste forms is provided in this Statement as additional background information.

waste management and the ultimate disposal of certain nuclear materials. The President in a Message to Congress on 12 February 1980 reiterated the role of the Department as the lead agency for the management and disposal of radioactive wastes (5). (A copy of his Message is attached as Appendix A.) Pursuant to this responsibility, the Department is conducting the National Waste Terminal Storage (NWTS) Program to develop a plan for the disposal of radioactive waste, to develop methods to isolate such waste and to identify appropriate sites for disposal facilities.

For interim storage of spent fuel, the Administration has proposed to Congress a bill which would give the Department authority to enter into contracts with NRC licensees to accept commercial spent fuel for storage and disposal at federally operated facilities (6). The Department is developing a program for prompt implementation of the bill, once enacted.

The Department and its predecessor agencies have been involved in the management of radioactive waste since 1944 when radioactive waste was first generated as a byproduct of national defense programs. The principal source of this waste has been the reprocessing of reactor fuel to recover fissionable materials for use in the Nation's defense program. Radioactive waste also has resulted from the production of components for weapons, laboratory experiments, and reactor operations (7).

The objectives of the Department's operating waste management program for safe interim storage are to ensure that (i) DOE radioactive wastes are handled, processed, stored, and disposed of safely; and (ii) interim storage of the waste is accomplished in a manner compatible with options under consideration for long-term management. The interim storage program has focused on providing multiple containment barriers to prevent releases of radioactive material to the environment, surveillance and monitoring of waste storage facilities and the surrounding environment, and maintenance and improvement of waste handling and storage facilities (8).

In the past 35 years, the Department and its predecessor agencies have accumulated thousands of man-years of experience in managing radioactive waste at various sites around the country. During this time, active health and safety programs have been maintained to reduce industrial and radiological accidents to levels as low as reasonably achievable. Accidents and releases of radioactive materials have occurred, but there have been no injuries to members of the public or serious environmental damage as a result of these operations.

The environmental impacts of the Department's waste management operations at various sites have been analyzed extensively. Environmental impact statements (EIS's) have been published for the Hanford, Savannah River, and Idaho sites (9-11). An EIS for the Oak Ridge site is being prepared. Numerous other environmental and safety analyses have been prepared for specific facilities and operations (12). The conclusions of the completed studies are that the Department's waste management operations present no significant radiological hazard or other adverse impact to employees, the public, or the environment. The Department's waste management program also has been assessed by independent organizations, such as the National Academy of Sciences and the General Accounting Office. These reviews, while identifying possible improvements, have shown that the Department's operations have not and do not present a significant hazard (13-15).

I.C POSITION OF THE DEPARTMENT OF ENERGY

The Department of Energy will demonstrate in this Statement of Position and throughout this proceeding that:

1. Spent nuclear fuel from licensed facilities ultimately can be disposed of safely off-site.

2. Disposal facilities will be in operation between 1997 and 2006, and the initial increment of off-site storage facilities can be in operation by 1983.

3. Spent nuclear fuel from licensed facilities can be stored safely either on-site or off-site until disposed of ultimately.

The issues facing the Nuclear Regulatory Commission in this rulemaking are what will be the disposition of spent fuel stored at the sites of operating power reactors and how questions about such disposition should be addressed in individual NRC licensing proceedings. In considering whether or not individual licensing proceedings should address the possibility that spent fuel might have to be retained at reactors beyond the expiration of their operating licenses, it is necessary for the Commission to determine whether or not off-site facilities for the disposal or storage of spent fuel will be available on a timely basis. In making this determination, it is necessary that the Commission find only that at least one method will be available for the off-site disposition of high-level nuclear wastes. The Department in this Statement assesses the technique by which spent fuel presently stored at utility reactor sites would be disposed of directly. In presenting its assessment of this technique, the Department is in no way suggesting a judgment of the potential suitability or nonsuitability of other techniques for treatment and disposal of radioactive wastes.

Because any program for off-site handling and storage of spent fuel must result ultimately in the safe and environmentally acceptable disposal of radioactive wastes, the Department in this Statement first discusses the issue of the ultimate disposal. This discussion addresses the technical basis upon which a determination can be made that nuclear waste from licensed facilities can be disposed of safely. This discussion considers throughout those issues requiring resolution prior to the successful disposal of these wastes. Also discussed is the program plan that will lead to the actual construction of disposal facilities. The Department does not attempt to prove that safe disposal of these radioactive wastes, with the required approval of the appropriate regulatory authorities, can be achieved today. Rather, the Department shows that such disposal can be achieved within reasonable times (which are specified) upon completion of its current research and development and site exploration programs.

Existing storage capacity for spent fuel being discharged by nuclear power reactors now in operation is being depleted. In view of the fact that a program to develop and begin operation of permanent disposal facilities, that will satisfy the detailed technical and administrative review of regulatory authorities may well require 15 years or more, it is necessary that in the interim, some temporary off-site storage of spent nuclear fuel be provided. The Department of Energy's proposed program would provide safe and environmentally acceptable off-site storage pending availability of disposal capacity. The Department has identified a need for a limited amount of off-site storage of spent fuel beginning in the early to mid-1980's and has determined that water basin storage is appropriate for Government-sponsored off-site storage facilities.

The Department of Energy's program for the management of high-level radioactive wastes is the program announced by the President of the United States in his Message to the Congress on 12 February 1980 (5). In his statement, the President outlined details of the program. With regard to waste disposal, he said:

> . . . for disposal of high-level radioactive waste, I am adopting an interim planning strategy focused on the use of mined geologic repositories capable of accepting both waste from reprocessing and unreprocessed commercial spent fuel. An interim strategy is needed since final decisions on many steps which need to be taken should be preceded by a full environmental review under the National Environmental Policy Act. In its search for suitable sites for high-level waste repositories, the Department of Energy has mounted an expanded and diversified program of geologic investigations that recognizes the importance of the interaction among geologic setting, repository host rock, waste form, and other engineered barriers on a site-specific basis. Immediate attention will focus on research and development and on locating and characterizing a number of potential repository sites in a variety of different geologic environments with diverse rock types. When four to five sites have been evaluated and found potentially suitable, one or more will be selected for development as a licensed, full-scale repository.
>
> It is important to stress the following two points: First, because the suitability of a geologic disposal site can be verified only through detailed and time-consuming site-specific evaluations, actual sites and their geologic environments must be carefully examined. Second, the development of a repository will proceed in a careful, step-by-step manner. Experience and information gained in each phase will be reviewed and evaluated to determine if there is enough knowledge to proceed with the next stage of development. We should be ready to select the site for the first, full-scale repository by about 1985 and have it operational by the mid 1990's . . .

With regard to storage of spent fuel, the President said:

> In contrast, storage of commercial spent fuel is primarily a responsibility of the utilities. I want to stress that interim spent fuel storage capacity is not an alternative to permanent disposal. However, adequate storage is necessary until repositories are available. I urge the utility industry to continue to take all actions necessary to store spent fuel in a manner that will protect the public and ensure efficient and safe operation of power reactors. However, a limited amount of government storage capacity would provide flexibility to our national waste disposal program and an alternative for those utilities which are unable to expand their storage capabilities.
>
> I reiterate the need for early enactment of my proposed spent nuclear fuel legislation. This proposal would authorize the Department of Energy to: (1) design, acquire or construct, and operate one or more away-from-reactor storage facilities, and (2) accept for storage, until permanent disposal facilities are available, domestic spent fuel, and a limited amount of foreign spent fuel in cases when such action would further our non-proliferation policy objectives. All costs of storage, including the cost of locating, constructing and operating permanent geologic repositories, will be recovered through fees paid by utilities and other users of the services and will ultimately be borne by those who benefit from the activities generating the wastes.

For the purpose of this rulemaking, finding confidence in the nation's ability to safely store and dispose of its radioactive waste requires finding that the Department of Energy's program will culminate in licensed storage and disposal systems, i.e., that the Department is able (i) to understand and address in its program the technical, social, political, and institutional aspects of waste management and (ii) to use the results of the program to develop licensed systems for the storage and disposal of spent fuel in the time frame required by national needs.

In order to demonstrate the Department's ability to understand and address the technical aspects of either temporary storage or permanent disposal of spent fuel, this Statement includes the following examination:

First, the various possible methods for storage or disposal of spent fuel are assessed, and the basis for selecting a preferred method is described. Then, each component of the preferred system is discussed individually. The factors important to its function are described, and each proposed or existing requirement for its performance is stated. Because the Department's program is to be judged on its present and future ability to develop components that meet these requirements, the Statement next describes the current ability of the program to meet such requirements. Where technical uncertainties concerning that ability remain, the Statement describes the research and development still to be performed and explains how the results therefore are expected to resolve these uncertainties.

This Statement shows that almost no technical uncertainties exist concerning safe and environmentally acceptable interim storage. Some technical uncertainties remain, however, in the disposal program. Uncertainties are accounted for by providing features in the mined geologic disposal program that lead to a relative insensitivity to uncertainties. This is achieved through the application of appropriate site suitability criteria to diverse geologic environments; the provision of redundant, independent, natural, and engineered features to retard the movement of radionuclides; and the application of conservative engineering practices. Uncertainties can be evaluated by bounding their potential impacts on system performance. Bounding analyses that take into account site suitability and redundant features provide current confidence in the future performance of disposal systems.

By applying a conservative step-by-step approach in the program, experience and information gained at each step will contribute to a reduction in uncertainties and provide a basis for proceeding to the next step. However, even after the research and development is completed, residual uncertainties, smaller than those which now exist, will still remain. The Department's program and the ability to provide engineered barriers in a disposal system afford sufficient flexibility to accommodate these residual uncertainties in systems for the safe storage and permanent disposal of radioactive waste.

To demonstrate the Department of Energy's ability to understand and address the social, political, and institutional aspects of waste management, the Department's program plans and management structures are presented. Also discussed is a description of both current and planned institutional arrangements that have been or will be established to enable the Nation to address radioactive waste management problems in a manner that promotes cooperative relationships between Federal and State governments. As is discussed later in this Statement, the recent initiative of the President of the United States in establishing a State Planning Council to strengthen intergovernmental relationships and help fulfill joint responsibilities to protect public health and safety is an important contribution to the development of the necessary institutional arrangements (16).

This Statement demonstrates that the technical and management aspects of the Department's radioactive waste management program are well developed. Important technical and institutional issues are understood and are being addressed so that licensable systems for storage and disposal of spent fuel will result. The comprehensive review of the Interagency Review Group on Nuclear Waste Management completed in 1979 and the strong support given to the program by the Administration, as reflected in the President's statement on 12 February 1980, have substantially strengthened the program.

Recognition of the vital importance of the management of radioactive wastes in the last several years has resulted in a significant growth in the Department's program, requiring the establishment of a broadly based management structure in the areas of both research and development and institutional arrangements. In some cases, the management structure to integrate these areas has been established only recently, and it is clear that continued attention to this will be required. However, the Department submits that the detailed attention that has been provided to this program at the highest levels of government will continue and will enable the program to proceed as outlined by Administration policy.

As a result of the extensive review and evaluation over the last 2 years, the Department's technical program for waste disposal has expanded greatly. The technical basis for the program has been substantially broadened. Additional scientific disciplines have been recognized as important and investigators from these fields are now active in the program. The examination of potentially suitable disposal sites has been diversified to encompass a variety of environments with diverse rock types. Extensive site selection work currently is being carried out in a number of different locations around the United States and work on several new regions is to begin shortly. The bases for decisions are now being established both internally by the Department of Energy and externally by the responsible regulatory authorities, namely, the Nuclear Regulatory Commission and the Environmental Protection Agency. The development of specific criteria and standards against which to measure the progress of the technical program and toward which responsible technical participants in the program can focus their efforts will provide additional impetus to the successful completion of the required technical work. Finally, using a conservative approach will ensure that whatever technical uncertainties might still remain will not diminish confidence in the successful operation of storage and disposal facilities meeting the required standards set forth by independent regulatory authorities.

The expansion of the current waste disposal program has built on work conducted for over 20 years. Investigation of geologic disposal as a means to provide safe isolation of radioactive wastes has been under way since 1957 when the initial recommendation that such disposal should be considered as a primary candidate was made by a committee of the National Academy of Sciences. Whereas many of these prior investigations may not have been structured strictly within the format now prescribed for the program, such preliminary work has been essential in developing an understanding of the parameters of importance and guiding the future direction of the program. It is now necessary to integrate the results of the prior investigations into the broader program that recently has been identified by the intensive examination of the Interagency Review Group and by the definition of program strategy by the President. This Statement shows that the integration of the previous work and the current strategy for the development of suitable repository sites can be achieved readily and that, indeed, the preliminary work from the last 20 years substantially contributes to the successful implementation of the current program strategy.

In this Statement, the Department shows that implementation of the waste disposal strategy will result in the establishment of operating geologic repositories in the time range of 1997 to 2006. The exact date of operation depends upon a number of variables, which will be determined only by the outcome of existing programs. If the examination of potential repository sites in a variety of geologic environments with diverse rock types indicates that a site in bedded or dome salt is prefer for the initial repository, the reduced construction time possible in salt and an assumption of licensing schedules recently forecast by the Nuclear Regulatory Commission staff could lead to the operation of a repository as early as 1997. On the other hand, if further examination indicates that a repository in hard rock (such as granite) would be preferable, construction in that medium would require more time before the operation of a repository could begin. Furthermore, if allowances are made for other uncertainties (such as the time required for licensing proceedings or for collection of more extensive preliminary data than currently planned prior to the licensing proceedings), the sequential imposition of these allowances on future forecasts could result in initial repository operation as late as 2006.

In view of these predicted dates of repository availability, the Department also is pursuing a program to provide temporary storage in off-site, or away-from-reactor, storage facilities. While the storage of commercial spent fuel is primarily a responsibility of the electric utilities, the Department's program for government off-site storage capacity will provide flexibility in the national waste disposal program and an alternative for those utilities unable to expand their own storage capacities (5).

This Statement describes the technical basis for findings that such off-site storage can be provided; that current program plans provide for the availability of storage facilities beginning in 1983; and that the provision of such facilities can accommodate storage needs until repositories become operational. The availability of these storage facilities will allow sufficient time for careful and prudent development of safe disposal facilities.

In summary, the President has established a national waste management program which is being conducted by the Department of Energy. The Department's waste management program has been significantly expanded and has incorporated the results of intensive review by several Federal agencies, Congressional committees, State officials, scientific groups, and the general public. Furthermore, the Environmental Protection Agency and the Nuclear Regulatory Commission are both actively engaged in developing a regulatory framework that will permit licensing of waste management facilities proposed by the Department.

The Department of Energy submits that the waste disposal program described in this Statement provides the basis for a finding that spent nuclear fuel from licensed facilities will be disposed of safely within a reasonable time. While there is no technical reason why storage at reactor sites cannot be continued for extended periods of time, it is recognized that shortage of storage capacity will require additional storage facilities off-site. The technical basis for construction of such off-site storage facilities for spent nuclear fuel and a capability to provide such facilities in the mid-1980's are available now. The Department of Energy therefore submits that spent nuclear fuel from licensed facilities can be both stored and disposed of safely off-site.

I.D SUMMARY OF THE DEPARTMENT OF ENERGY'S DETAILED PRESENTATION

As previously described, the Department of Energy's presentation in this statement begins with disposal of unreprocessed spent fuel. The discussion of disposal is presented first because the requirements for spent fuel storage are related directly to the availability and capacity of disposal facilities. The amount of spent fuel which can be disposed of and the schedule for disposal will affect the storage requirements. After discussion of disposal, storage is reviewed. Finally, this Statement shows how the disposal system and the storage system will be integrated and provides examples of how the two systems relate to each other.

The Department's discussion of disposal is divided into two parts: First, this presentation explores the technical objectives and requirements of a disposal system and demonstrates how those can be met. This portion of the discussion also addresses uncertainties remaining in the program and shows the effect of uncertainties on the performance of the system. The Department's program for the establishment of an operating geologic repository for the emplacement of spent nuclear fuel is then described. The same order of presentation is followed in the discussion of storage.

I.D.1 Technical Basis for Disposal

The goal of safe disposal is the effective isolation of radionuclides from the environment in a safe and environmentally acceptable manner.

The Nuclear Regulatory Commission will judge the adequacy of any high-level waste disposal system based upon predetermined criteria and requirements. The ability of the Commission to reach a satisfactory conclusion regarding the safety aspects of an application will be directly affected by the manner and degree of demonstrated conformance with such criteria and requirements. Although the Department is using conservative criteria and methods, there is a need to ensure that the Department's approach will be compatible with that required by the Commission and be amenable to timely Commission review. Because final NRC regulations are not yet available, the Department has developed proposed generic performance objectives, based upon a review of a wide variety of publications on the subject. The Department proposes that these will be similar to those that can be expected to be promulgated by the Commission and suggests that they be used in this proceeding to evalute the Department's program. The generic performance objectives which the Department is proposing should be met by any disposal system are summarized as follows:

1. Containment should be virtually complete during the period dominated by fission product decay.

2. Isolation from the accessible environment should be effective for at least 10,000 years, and reasonably foreseeable events should not produce consequences greater than normal variation in background radiation.

3. The operational phase of a waste disposal system should be as safe as for other nuclear fuel-cycle facilities.

4. Environmental impacts should be mitigated to the extent reasonably achievable.

5. Conservative design and evaluation should be applied to waste disposal systems to compensate for any residual uncertainties.

6. Acceptable performance should be based on methods reasonably available and should not depend upon continued maintenance or surveillance for unreasonable times into the future.

7. Concepts selected for implementation should be independent of nuclear industry trends and compatible with national policies.

To implement these objectives for any alternative disposal method, the Department has adopted a requirement for conservatism in design and operation. This requirement is accomplished by (i) a step-wise approach to continually reassess the state of knowledge and to assure that designs and plans are supported by the best and latest data; (ii) a multibarrier approach in which waste is isolated from the biosphere by a series of relatively independent and diverse barriers that would not be subject to common failure; and (iii) use of design and operating margins to compensate for uncertainties in design and knowledge of natural systems.

Following discussion of the basic objectives for disposal and the conservative approach to achieve the overall goal, the Statement outlines a number of disposal alternatives and explains the basis for selecting the method of disposing of nuclear waste in mined geologic repositories. This method has been adopted as an interim planning strategy, i.e., not final but subject to change until the completion of environmental reviews now being conducted by the Department. Prior to the completion of these environmental reviews, no decision or commitments that would foreclose alternatives can be made.

Mined geologic repositories appear most likely to meet all of the proposed objectives. It is believed that locations within the Earth's crust whose primary change mechanisms require geologic time periods to occur and which appear to provide negligible hydrologic transport potential are suitable for the permanent isolation of nuclear waste.

The alternatives of subseabed disposal or disposal in very deep holes appear more amenable to being assessed with reasonably available methods, but questions remain which must be addressed. They do, however, appear sufficiently promising that continued examination to assess their potential for later development is warranted.

The mined geologic disposal system will incorporate both natural and man-made systems which are composed of 3 major subsystems: the natural system associated with the site, the waste package and the repository. Together they will provide multiple barriers between the emplaced waste and the human environment. The natural system, the natural geologic and hydrologic features of the repository site, as well as the remoteness of the repository (in terms of depth below the surface and distance from water supplies), provide barriers for isolating nuclear waste from people and their environment. Engineered barriers incorporated in the waste package and repository system provide containment of the waste delaying the time and retarding the rate of release of radionuclides into the far-field environment. Prior to repository closure, the waste package and waste form will aid in protecting both the repository work force and the general public by containing the waste and limiting the potential for its dispersal if the waste package is breached.

The natural system for waste isolation consists of the repository host rock, surrounding geologic formations, and the associated hydrologic environment. It is discussed in the context of a near field and a far field. The near field provides both containment* and isolation** for the emplaced

---

*Containment means confining the radioactive wastes within prescribed boundaries, e.g., within a waste package.

**Isolation means segregating waste from the accessible environment (biosphere) to the extent required to meet applicable radiological performance objectives.

waste: containment by minimizing the likelihood that circulating ground water will contact the waste package, and isolation by ensuring that any migration of radionuclides will be very slow. The prime function of the far field is to ensure that, if radionuclides were released from the near field, ensuing migration to the biosphere would be of sufficient duration to satisfy the second generic performance objective set forth previously.

The current status of natural systems investigations can be summarized as follows:

1. The scope of technical information required for evaluating natural systems and the role that natural systems can play in providing barriers for containment and isolation are known.

2. Required characterization techniques are available; many represent the state of the art.

3. The need for additional improvement in predicting performance of fractured, and perhaps water-bearing, rock masses has been recognized.

4. Site identification programs are being conducted in a number of regions and host rocks, including basalt, granite, shale, salt, and tuff; some are well advanced.

5. Investigations to date strongly suggest that acceptable natural systems exist that will meet the performance objectives.

6. The diversity of media under evaluation, the large number of potentially suitable sites contained in the areas and regions being studied, and the NWTS Program's ability to successfullly screen for sites using criteria and the available performance assessment techniques will result in identifying, qualifying, and licensing repository sites.

The discussion then turns from the natural systems of mined geologic disposal to the man-made systems. These systems are made up of three basic functional components: the waste package system, the repository system, and human intrusion barriers.

The waste package system includes everything man places in the repository waste emplacement hole, e.g., the waste form, canister, overpack, and backfill. These various package system components will be used to reduce overall technical uncertainties by virtue of their conservative engineering design and by providing barriers in addition to those provided by the host rock and surrounding strata.

Extensive testing and development studies have been underway for several years on the waste package and specific components of the package, e.g., waste forms. While these studies are not complete, results to date indicate that the waste packages can be designed and fabricated to meet the previously stated objective to provide virtually complete containment of the wastes during the period dominated by fission product decay.

The second component of the man-made systems, the repository system, provides for the receipt, inspection, transfer to the underground emplacement, and containment after closure of radioactive waste. The system must also contribute to the long-term isolation of the waste by limiting adverse impacts on the natural isolation system during development of the repository, and to the extent possible, by enhancing isolation through the use of engineered barriers. Thus, activities involving (i) the introduction of heat and radiation generated by the waste, (ii) the excavation of underground disposal areas, and (iii) the introduction of penetrations (such as exploration boreholes and repository shafts), will be conducted so as to minimize any adverse impacts on the integrity of this natural isolation system. Thermal impacts will be minimized by limiting the thermal loading and thus the temperatures in the repository. Migration of radionuclides will be restricted by the use of sorptive backfill materials. Impacts of repository excavation on structural stability will be limited by using low extraction ratios, highly-developed and widely-applied excavation techniques and the backfilling of rooms and tunnels.

Progress in ongoing research and development programs indicates that satisfactory designs and materials will be available for sealing of penetration.

Human intrusion barriers, the third component of man-made systems, are provided so that the waste will be unaffected by future activities of man. These barriers are intended to meet two objectives: (i) to reduce the likelihood of human-induced releases, and (ii) to mitigate the consequences if human-induced releases occur. Although work is just beginning in this area and there is much to be learned, it is reasonable to conclude that (i) the likelihood of future human activities of a nature which could adversely affect the integrity of the repository can be reduced to an acceptably low probability through the use of appropriate protective measures, and (ii) the impact of any such future activities, were they to occur, could be adequately mitigated by the multiple natural and man-made barriers included in waste disposal systems.

The Statement then moves to a demonstration of the methods of safety analysis in use and to be used to determine that the requirements identified will be met by the expected performance of repository components. The discussion of safety is covered under two topics: (i) long-term performance assessment, and (ii) operational phase performance assessment.

Performance assessment methods have been developed to analyze the disposal system after the waste has been emplaced and the repository has been sealed. These methods analyze the combined effects of several phenomena which might affect the disposal system: natural events and processes, human actions, and impacts exerted by the waste and the repository. Three kinds of future conditions can then be identified:

1. Conditions under which radionuclide releases from the package would not occur.

2. Conditions under which radionuclide releases from the waste package occur but radiation doses are not received by people.

3. Conditions under which released radionuclides would deliver radiation doses to peple.

Because the disposal system contains components which have complex interactions with one another and because its performance for long time periods must be predicted, it is necessary to use mathematical models to analyze the system. The discussion describes how those models are developed, verified, and applied to gain confidence that the long-term performance of the disposal system will be acceptable.

The development and verification of models of single phenomena required for analyzing the long-term performance of geologic disposal systems are well advanced, and these models can now be routinely used. Even so, the development and particularly the verification are continuing. The development and verification of models which analyze several phenomena together are moderately advanced, and some of these models can now be routinely applied. Laboratory, bench-scale and in situ tests are underway or planned to assist in verifying modeling predictions.

The use of these continually improving models, along with the continually improving body of experimental data, will permit the performance assessments to be done more completely and with increased confidence. These assessments will be important in site selection, system design, and licensing.

The models have been used to assess the performance of disposal systems and have demonstrated that they can analyze the important phenomena. Because the development and verifications of models and the gathering of data describing sites are incomplete, these assessments have used conservative data derived from laboratory and field measurements. They have demonstrated that the models have been developed sufficiently for use with complex systems.

These studies have also predicted the consequences of releases of radionuclides from repositories in the far future. The vast majority of possible disposal-system conditions would not deliver any measurable doses to people. Some possible but unlikely phenomena, such as ground water flow directly through repositories, could deliver radiation doses that would be a fraction of the doses delivered by natural background radiation. The analyses performed to date give no indication that a geologic disposal system, designed and constructed according to the requirements described in this Statement, cannot isolate radioactive waste safely.

Assessing the performance of the repository during the operational phase does not require the special methods of long-term performance assessment because the operations are at least similar to those in other common systems. For example, excavation of storage rooms and packaging and handling of waste containers are not too different from operations currently being performed. Adequate methods required for safety analyses are currently available. Satisfactory design, construction and operation can be achieved with repository performance. The operational phase activities can be shown to be comparable in safety to those of existing licensed nuclear fuel cycle facilities.

Since the mid-1950's many geologic and environmental studies have been conducted to provide the technical and scientific basis for the design, construction, and operation of deep, underground repositories for radioactive wastes. A broad spectrum of agencies and organizations has sponsored this research. Results indicate that a mined geological repository can be build and operated safely with minimal effects on people and their environment.

In this statement, the Department describes a program which has:

1. Proposed specific proposed performance objectives.
2. Adopted a conservative approach to ensure the objectives will be met.
3. Selected mined geologic disposal as an interim planning strategy.
4. Identified characteristics and requirements necessary for mined geologic disposal.
5. Developed specific criteria for site qualification, established conceptual designs and identified elements warranting more research.
6. Completed performance assessments indicating that carefully designed repositories in properly selected sites will meet the proposed performance objectives.

The Department, therefore, submits that a mined geologic disposal system can meet the goal of providing the effective isolation of radionuclides from the environment because:

1. Containment will be virtually complete during the period dominated by fission product decay.
2. Isolation will be effective for at least 10,000 years.

3. The operational phase of the waste disposal system will be as acceptably safe as the operation of other nuclear fuel cycle facilities.

4. There are no unreasonable environmental impacts.

5. Conservative design and evaluation will compensate for residual uncertainties.

6. Acceptable performance is based on a reasonably available level of technology and is not dependent upon continued maintenance or surveilance for unreasonable times into the future.

7. Implementation is independent of the size of the nuclear industry and of the resolution of fuel-cycle or reactor design issues and is compatible with national policies.

I.D.2 Program for Establishing Mined Geologic Repositories

The program for establishing an operating geologic repository is presented next. This presentation begins with a discussion of the Department's management organization and the major decisions required to implement the geological disposal option.

The Department's program is focused on developing repositories that will be available in an appropriate time frame and at a reasonable cost. To accomplish this goal, the Department has put into place a management organization to address and resolve the technological, societal, economic, regulatory, and institutional factors which could have impacts on the timing and cost of the National Waste Terminal Storage (NWTS) Program. The Department has established an organization consisting of headquarters and field office personnel supported by over 2,000 professional employees of contractors to implement the NWTS Program. This arrangement brings experience from a broad spectrum of professionals ranging from geoscientists and mechanical engineers to sociologists and political scientists.

The schedule for implementation of the geologic disposal option depends on the following major decisions:

1. Site selection with State consultation and concurrence.

2. Licensing for construction of the repository.

These decisions are the focus of technical activities which must be conducted in cooperation with those agencies, organizations, and individuals outside of the Department which are part of the decisionmaking process.

As directed in the President's statement of 12 February 1980, the Department intends to identify candidate sites at several locations and in different geologic media before recommending a specific site for the first license application. The selection of candidate sites will be based on a systematic process which considers all applicable factors and is conducted with involvement of State and local officials and the public. The Department's program leading to selection of a site consists of three phases:

1. Site exploration and characterization.
2. Detailed site characterization.
3. Site selection.

Regional and area characterizations are now under way for various geologic media, including dome salt and bedded salt, basalt flows and volcanic tuff. Efforts have been initiated to identify regions containing other media and other geohydrologic systems. The Department plans to identify at least four sites with diverse rock types by mid-1985. The DOE approach includes consideration of regulatory factors, environmental factors, the necessity of achieving public acceptance, and the need to meet site qualification criteria.

The involvement of the States in the radioactive waste management program was addressed by the President in his 12 February 1980 statement, in which he announced the establishment of a State Planning Council to advise the Executive Branch and work with Congress on repository planning and siting, construction, and operation of facilities (16). Further, the President provided for the relationship with the States to be based on the principle of consultation and concurrence. In compliance with the President's statement, the Department's program seeks to take into account State and local needs and concerns. A process is being developed that provides for cooperative Federal, State, and local government decisionmaking concerning identification of candidate sites and selection of sites for license application.

The second major decision which can affect the repository schedule is licensing. The Commission has the statutory authority to license facilities used primarily for the receipt and storage of high-level radioactive wastes resulting from activities licensed under the Atomic Energy Act of 1954. Accordingly, the availability of regulatory procedures and requirements can have an impact on schedule of a disposal system. Before a license application is submitted, the Department expects to consult with the Commission staff in regard to field exploration activities so that the integrity of a potential site is not adversely affected.

The existing knowledge of licensing requirements as obtained from draft and proposed rules and communication with regulatory agencies has allowed a forecast of the effects licensing activities will have on the schedule. Based on the existing information, estimates can be made of the amount of time which will be required for licensing. The attainment of program milestones will depend on the exact form of the procedural rules promulgated by the Commission and on the manner in which the Commission will process future license applications.

In addition to the major decisions discussed above, several significant factors also can influence the timing and schedule of a repository:

1. Implementation of National Environmental Policy Act.
2. Cooperation of multiple Federal agencies.
3. Land acquisition activities.
4. Availability of expert staff.
5. Logistics and administration.
6. Design and construction time.
7. Initial operation period and backfill.
8. Technology development.

The National Environmental Policy Act of 1969 (NEPA), as implemented by the regulations of the Council of Environmental Quality (CEQ) and the DOE guidelines, requires that potential environmental consequences be considered in Department planning and decisionmaking. Using DOE guidelines and the CEQ regulations, the Department has developed a draft NEPA implementation plan for the mined geologic disposal interim strategy which is integrated with the overall Department planning and decisionmaking framework. The environmental impacts of all reasonable alternatives will be considered at each stage of the decisionmaking process.

The Presidential statement of 12 February 1980 emphasizes the commitment to provide for safe disposal of radioactive wastes with support from all agencies within the Administration. In its role as lead agency for the management and disposal of radioactive wastes, the Department is preparing, with cooperation of other cognizant Federal agencies, a detailed National Plan for Nuclear Waste Management to implement the Presidential policy guidelines and IRG recommendations. The program content now includes many activities specifically recommended by the Interagency Review Group so that other agencies will support the Department activities where required. The ability to draw on the resources of such organizations and to obtain meaningful comments and direction will greatly enhance the ability of the Department to meet major milestones.

Where the Department does not already own or control a proposed repository site, the acquisition of the real property for the repository must be considered in the schedule. Non-Federal land can be acquired by the Department for a repository site by purchase or condemnation following procedures already established. Federal property controlled by other agencies may be acquired by transfer to the Department following procedures established by the General Services Administration or by the Department of the Interior.

Skilled technical personnel will be required in the site exploration and characterization phase of the program, in the development of necessary technology, and in the design and construction phase of the repository. The design and construction expertise required to build a geologic repository is currently available in the United States. Operating expertise will be available by the time the repository is ready for waste emplacement. The schedule for implementing the repository option should not be affected by the need for lead time to develop any needed expertise.

Evaluation of logistic and administrative factors shows that with proper planning, schedule slippage can be minimized. Many factors, such as procurement delays and strikes and labor disputes, can influence the construction schedule. Prudent planning takes these factors into account to minimize such delays. Normal design and construction scheduling practices will ensure that the repository will be developed in the projected time frame.

Although not strictly associated with the schedule to bring a repository into operation, the waste retrieval period and start of backfill both have an impact upon the time when a repository can become fully operational. During the initial repository operation period, the Department will verify the predictive capability of methods used to apply early geologic test data to the specific site and design configuration and will verify that no unforeseen phenomenon associated with actual waste emplacement is observed. Ample latitude is provided for methodical, step-wise development including testing and evaluation. A high level of confidence concerning the integrity of the operation will be attained before backfilling will commence. Should retrieval of waste be necessary following the initiation of backfilling, waste management plans include rerouting the wastes to other facilities.

The Department is proceeding with a systematic program to develop the needed technology on a timely basis. A detailed discussion of the base technology program is presented in the Statement.

On the basis of estimates described in more detail in the presentation, the Department projects that the range of startup dates for the first repository is 1997-2006.

I.D.3 Technical Basis for Storage

The Department's Statement next discusses the technical basis for the conclusion that spent fuel can be stored in a safe and environmentally acceptable manner until disposal facilities are available. The Statement (i) describes the alternative methods and technology for storage of spent fuel, (ii) reviews the performance requirements of storage facilities and how these requirements are met, and (iii) discusses the background of experience which leads to the selection of water pool storage as the preferred method. The abundant evidence shows that the adequacy and safety of extended storage of spent fuel in a water pool environment can be demonstrated today. The following points summarize the basis of this conclusion:

1. The technology of water pool storage of spent fuel is not only available but is well established through more than 30 years of work at government and industrial facilities. Dry storage of spent fuel by several different techniques has been the subject of a significant level of research, development, and demonstration, and promises to be a technically viable alternative to water pool storage. Thus, there are a number of technically suitable alternative methods of spent fuel storage in existence at the present time.

2. The regulatory framework, industry standards, and design requirements for the water pool storage of spent fuel currently exist.

3. The licensing of water pool storage of spent fuel has been routinely practiced by the Commission and its predecessor agency for nearly 20 years and is being practiced at the present time.

4. Zircaloy-clad spent fuel has been stored under water for periods of up to 20 years, and stainless steel-clad fuel has been so stored for periods up to 12 years, with no evidence of degradation as a result of such storage. Studies of the corrosion aspects of water pool storage indicate that there are no obvious degradation mechanisms which operate on the cladding at rates which would be expected to cause failure in the time frame of 50 years or longer. Moreover, in the unlikely event that severe deterioration of the cladding were to develop, the spent fuel could be encapsulated to provide the necessary integrity for indefinite storage.

Because much of the experience in handling and storage of spent fuel has been gained at reactor sites and much of the technical data presented and discussed in this Statement was acquired in studies at reactor storage pools, there is no reason to doubt the technical adequacy of existing and planned reactor storage pools. Accordingly, the Department submits that continued storage of spent fuel at reactor sites would be acceptable, even if such storage should be required for a period beyond the expiration of the reactor operating license. This conclusion is based on the following considerations:

1. Continued water basin storage at reactor sites involves the same kinds of considerations as those set forth above with respect to:

    a. Performance of fuel under extended storage conditions.

    b. The benign character of the storage activity relative to radioactivity releases, radiation exposures to plant workers and the public, accident evaluations, and other safety aspects.

    c. Waste generation during the course of storage.

    d. Facility requirements.

2. Facilities in existence and those under construction at reactor sites are designed and constructed to more rigorous standards than would be required under proposed 10 CFR 72.

3. The environmental impact of continued storage of spent fuel at reactor sites has been evaluated by the Nuclear Regulatory Commission and found to be acceptable. No time-dependent factors adversely affecting long-term safety of such storage have been identified.

4. Although there is some interdependence between the spent fuel pool and reactor operation at reactor sites, this interdependence is limited to the supply of utilities and waste management services. The fuel storage operation depends on the reactor plant utility system for steam, cooling water, deionized water, and handling of low-level wastes, as well as for certain personnel services such as health physics, and safety. All of these could be continued relatively easily following shutdown of the reactor.

5. There is no technical reason why spent fuel cannot be stored at the reactor site after reactor operation ceases.

6. Such continued storage would remain under NRC licensing authority.

These factors regarding the confidence in water pool storage generally are applicable to away-from-reactor (AFR) storage systems. Therefore, the Department submits there is sufficient technical information provided in this Statement to support a finding that AFR spent fuel storage facilities can be built and operated in a manner which is both safe and environmentally acceptable, and which otherwise meets the performance requirements established by the Commission's regulations. The information presented by the Department demonstrates that large AFR spent fuel storage facilities can be constructed and operated to meet necessary safety requirements with a minimum impact on the environment and in compliance with applicable regulations. These conclusions are supported by the generic environmental impact statement on spent fuel storage prepared by the Nuclear Regulatory Commission and the draft environmental impact statement on spent fuel policy prepared by the Department.

I.D.4 Program for Providing Away-from-Reactor Storage Facilities

The Statement next considers the need for AFR storage capacity for spent fuel and the management plan which has been established and is being implemented by the Department to ensure that the AFR storage capability is available when needed.

The Department estimates that AFR spent fuel storage facilities can be made available commencing as early as 3 years after Congressional authorization and that the necessary AFR storage capacity can be maintained until geologic repositories are available for disposal of spent fuel. The Statement demonstrates the following to support this conclusion:

1. The near-term (through 1990) needs for AFR storage capacity can be satisfied by acquisition and expansion of the storage capacity of existing facilities. The longer term needs for storage can be satisified by building additional AFR storage facilities to supplement existing and expanded facilities. Needed capacities could be made available in existing storage facilities within approximately 3 years after Congressional authorization and could be made available in new AFR storage facilities within 95 months after such authorization.

2. The Department has established a management organization for the development and operation of AFR storage facilities.

3. There is sufficient information available on which to base selection of new sites for AFR storage, and the Department has embarked on a program designed to seek State involvement in the selection of proposed sites.

4. Finalization of new regulations pertaining to AFR storage of spent fuel is under way, and technical ability to meet such licensing requirements exists.

5. Legislation has been submitted to Congress for authorization to acquire the necessary AFR spent fuel storage capability.

6. Analysis of the requirements for AFR storage capacity in the near term, and of the steps which must be taken to comply with National Environmental Policy Act and other licensing requirements, indicates that the necessary capacity can be provided on a timely basis.

I.D.5 Integrated Operation of the Storage and Disposal Systems

The Department's presentation finally considers the integration of the mined geologic repository program and the AFR storage program to demonstrate that an overall waste management program exists which is capable of handling, storing, and disposing of the spent fuel. While studies to optimize the integrated system have not been completed, an example spent fuel management scenario is analyzed. Variables considered include the capacity, receiving capability, and date of availability of geologic disposal facilities, AFR storage availability and required capacity, and the transportation logistics for moving spent fuel. It is shown that the combined system of repositories and AFR's will provide great flexibility to meet the need to balance technical conservativism, regional needs, and reactor operation requirements. There are sufficient commercial organizations to provide the required shipping casks and services for the transport of spent fuel to AFR storage facilities and to repositories as needed.

The overall costs of the waste management activities are outlined, The Department submits that the impact of the cost of waste management program on the cost of electricity to the consumer will be small.

I.D.6 Conclusions

Based upon the information contained in this Statement, the Department in its conclusion submits that the Commission must find that:

1. Spent nuclear fuel from licensed facilities can be disposed of in a safe and environmentally acceptable manner;

2. The Federal Government's plans for establishing geologic repositories are an effective and reasonable means for developing a safe and environmentally acceptable disposal system;

3. Spent nuclear fuel from licensed facilities can be stored in a safe and environmentally acceptable manner on-site or off-site until disposal facilities are available;

4. Sufficient additional storage capacity for spent nuclear fuel from licensed facilities will be established; and

5. The disposal and interim storage systems for spent nuclear fuel from licensed facilities will be integrated into an acceptable operating system.

As noted in the Notice of Proposed Rulemaking (1), the Commission has indicated that it will use its findings in this proceeding to determine to what extent issues of on-site storage of spent nuclear fuel need be considered in individual facility licensing proceedings. Having made these five findings, the Commission should promulgate a rule providing that the safety and environmental implications of spent nuclear fuel remaining on site after the anticipated expiration of the facility licenses involved need not be considered in individual facility licensing proceedings.

REFERENCES FOR PART I

(1) 44 Fed. Reg. 61,372 (1979)

(2) Minnesota v. NRC, 602 F.2d 412 (D.C. Cir. 1979)

(3) 42 Fed. Reg. 34,391 (1977), pet. for rev. dismissed sub nom. NRDC V. NRC, 582 F. 2d 166 (2d Cir. 1978)

(4) In the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste, First Prehearing Conference Order, Docket No. PR-50,51 (NRC), February 1, 1980.

(5) Presidential Message to the Congress, February 12, 1980, "Comprehensive Radioactive Waste Management Program," Weekly Compilation of Presidential Documents, Vol. 16, No. 7

(6) The Spent Fuel Act of 1979, H.R. 2583 and H.R. 2611; S.792 and S.793

(7) U.S. Department of Energy, ASNE, Nuclear Waste Management Program Summary Document FY 1981, p. III-3, January 1980

(8) Ibid., p. III-7

(9) U.S. Energy Research and Development Administration, Environmental Impact Statement, Waste Management Operations, ERDA-1536, Idaho national Engineering Laboratory, Idaho Falls, ID, September 1977

(10) U.S. Energy Research and Development Administration, Environmental Impact Statement, Waste Management Operations, ERDA-1537, Savannah River, SC, September 1977

(11) U.S. Energy Research and Development Administration, Environmental Impact Statement, Waste Management Operations, ERDA-1538, Richland, WA, December 1975

(12) U.S. Energy Research and Development Administration, Environmental Impact Statements, ERDA-1515, -1520, -1521, -1527, to -1532, -1536, -1538, -1551, -1553

(13) Comptroller General, Nuclear Energy's Dilemma: Disposing of Hazardous Radioactive Waste Safely, September 1977

(14) Comptroller General, The Nations' Nuclear Waste--Proposals for Organization and Siting, June 1979

(15) National Academy of Sciences, Radioactive Wastes at the Hanford Reservation, A Technical Review, Washington, D.C., 1978

(16) Executive Order 12192 of February 12, 1980, "The State Planning Council of Radioactive Waste Management," 45 Fed. Reg. 9727 (February 13, 1980)

II TECHNICAL BASIS FOR CONFIDENCE THAT SPENT FUEL CAN BE DISPOSED OF IN A SAFE AND ENVIRONMENTALLY ACCEPTABLE MANNER

In his statement of 12 February 1980 (1), the President said:

> Our primary objective is to isolate existing and future radioactive waste from military and civilian activities from the biosphere and pose no significant threat to public health and safety. The responsibility for resolving military and civilian waste management problems shall not be deferred to future generations. The technical programs must meet all relevant radiological protection criteria as well as all other applicable regulatory requirements. This effort must proceed regardless of future developments within the nuclear industry--its future size, and resolution of specific fuel cycle and reactor design issues.

The Department of Energy's National Waste Terminal Storage (NWTS)* Program is designed to meet the objectives set forth by the President and the Interagency Review Group (IRG) (2, 3) and to develop licensable** high-level waste (HLW) disposal systems during this century. The technical aspects of the NWTS Program are discussed in detail in this part and are demonstrated to be in conformance with the President's program and the IRG recommendations.

For the purpose of this rulemaking, finding confidence in the nation's ability to safely dispose of its radioactive wastes requires finding that the NWTS Program will culminate in licensed waste disposal systems, i.e., that the Department is able (i) to understand and address the technical, social, political, and institutional aspects of waste management; and (ii) to use the results from its program to develop licensed systems for the disposal of spent fuel in a time frame which is responsive to national needs.

---

*NWTS is the designation of the Department's program for the disposal of high-level nuclear waste.

**In this context, licensable means consistent with regulatory requirements set forth by the Nuclear Regulatory Commission and the Environmental Protection Agency.

This Statement shows that the NWTS Program is designed to understand and address the technical, social, political, and institutional concerns important to waste management. In accordance with the President's 12 February 1980 statement, specific HLW disposal facilities will not be proposed by the Department for licensing for several years. Neither the Nuclear Regulatory Commission nor the Environmental Protection Agency has officially adopted specific regulations for waste disposal systems. In the interim, to ensure that the NWTS Program focus will result in licensable systems, the Department maintains continued awareness of regulatory trends by exchanging information in meetings with the NRC and EPA staffs and by reviewing and providing comments on proposed regulations prepared by those agencies. The Department has developed proposed HLW disposal system performance objectives to help focus the NWTS Program until specific regulatory requirements are available. These objectives provide an appropriate means of measuring the technical sufficiency of the NWTS Program in the context of this rulemaking.

Part II of this Statement demonstrates the technical sufficiency of the NWTS Program. It begins with a discussion of proposed performance objectives developed by the Department for the NWTS Program. These objectives provide program focus and direction pending the development of regulatory requirements. The analysis of disposal technology opens with a description of and rationale for the Department's commitment to a conservative approach to system development and operation. The elements of that conservative approach include a step-wise approach, a multibarrier system, and a provision for design and operating margins.

The remainder of Part II analyzes the NWTS disposal technology in the context of those objectives. Chapter II.B presents alternative disposal methods and analyzes the potential of each for meeting the NWTS objectives. Chapter II.B also presents the rationale for the selection of mined geologic disposal as an interim planning strategy.

Chapter II.C describes the basic features of a mined geologic disposal system to provide the reader a basic understanding of how such a disposal system will work. Chapters II.D, II.E, and II.F then present detailed technical discussions of a mined geologic disposal system.

The technical analyses presented in II.D through II.F demonstrate how mined geologic disposal, through both natural and man-made systems, is expected to achieve the objectives set forth for the program. Chapter II.D explains how the natural surroundings of a repository are expected to help meet the stated objectives and describes the potential impacts the natural surroundings will have on the repository. The man-made systems, which include engineered barriers, and their functions in a mined geologic disposal system, are described in II.E. The natural and man-made systems are expected to perform in concert to meet requirements set forth for the disposal system.

Since HLW disposal systems will be required to function far into the future without active assistance from man, the ability to assess and predict long-term system performance is a key factor in determining licensability. Chapter II.F details the methods, including mathematical modeling, used to perform such assessments. This discussion includes not only those methods presently available, but also assessment and prediction methods under development or undergoing refinement. Chapter II.G summarizes the information presented throughout Part II.

II.A PROPOSED HIGH-LEVEL WASTE DISPOSAL SYSTEM OBJECTIVES

Section II.A.1 discusses the objectives proposed by the Department to focus the development of its waste management systems. These objectives are drawn from diverse sources. Within the context of this rulemaking, these objectives are proposed as a basis for assessing the technical adequacy of the NWTS Program and of the systems that will result from its implementation, recognizing that, ultimately, such systems will be subject to NRC and EPA regulations and NRC approval. Section II.A.2 discusses the approach to conservative system development and operation used in the NWTS Program. The objectives and conservative approach described in this chapter allow sufficient flexibility within the research and development program to permit compliance with NRC and EPA regulations when they are adopted.

II.A.1 General Performance Objectives For the Safe And Environmentally Acceptable Disposal of High-Level Radioactive Waste

II.A.1.1 Background

Regulations promulgated by the Nuclear Regulatory Commission and the Environmental Protection Agency will provide guidance to the NWTS Program as the development of HLW disposal systems progresses and system details increase. The Commission has described its role and that of the Environmental Protection Agency as follows (4):

> NRC has licensing responsibility with respect to certain high-level waste repositories to assure protection of the public health and safety, taking into consideration also all impacts on the environment.
>
> EPA has the responsibility to establish generally applicable environmental radiation protection standards for areas outside the boundaries of nuclear facilities. NRC must devise its regulations in a manner that will meet requirements set by EPA.

The Commission and the EPA have published in the Federal Register procedural requirements and general criteria, respectively, for HLW disposal. The EPA has received comprehensive comments on its criteria, both through an interagency working group and from the public (5). The EPA staff has indicated informally in discussions with the NWTS Program staff that a general standard for HLW disposal will be issued for public comment during the Spring of 1980. The Commission in 1979 published the procedural portion of a proposed regulation for HLW disposal which addresses the proposed NRC approach to the regulation of HLW disposal facilities (6).

Although specific technical criteria from NRC and EPA would be useful at this time in directing the NWTS Program, they will not be critical to the conduct of the program until detailed waste disposal system designs are being developed. Frequent interactions among the Department and the NRC and EPA staffs provide a means for the NWTS Program to be responsive to potential regulatory concerns. Similarly, technical data from the NWTS Program will be of value to the Commission and the EPA in developing their technical regulations. Information from proposed EPA and NRC regulations is being utilized in planning the NWTS Program and consideration has been given to such information in the development of the NWTS performance objectives which follow (II.A.1.3).

The Commission staff recognized the need for NWTS objectives(4):

> DOE must determine the considerations it considers important to investigating and selecting repository sites. We will call these site selection criteria.

Recent emphasis on the man-made components of waste disposal systems and the diversity of disposal concepts which require consideration by the Department (see II.B) warrants more broadly based objectives than those related specifically to siting. Accordingly, the NWTS performance objectives are broad and generically applicable to a spectrum of alternative disposal methods.

The NWTS objectives are structured to allow adequate flexibility to meet specific regulatory requirements at the licensing stage. In structuring its objectives, the Department has placed emphasis on Presidential and IRG objectives and has used information generated by numerous studies, public meetings, task forces, working groups, and other appropriate forums, as well as information derived from draft and proposed regulations. The NWTS objectives are not intended to negate the need for NRC and EPA regulations but merely to provide interim guidance until comprehensive final regulations can be issued.

II.A.1.2 Information Relative to High-Level Waste Disposal Objectives

In light of the President's recent statement, the IRG findings and recommendations, and the large amount of information presented in technical meetings, regulatory workshops, and other related forums, a number of points have been made relative to HLW management. Pertinent points considered by the Department in structuring its objectives are as follows:

1. Waste should be isolated* so as to pose no significant threat to the public health and safety (1-3, 7).

2. Although zero release of radioactivity cannot be assured, potential releases should meet relevant radiological protection criteria and be reduced to the lowest practicable levels (1-3).

3. Radiological consequences should be maintained within the level of variations in natural background radiation associated with geographic location and domestic activities (8, 9).

4. Containment** and isolation are most important during the first few hundred years when the decay of short and intermediate-lived wastes is most rapid and radiation and heat levels are highest (8).

5. The performance of waste disposal systems may be compared to the natural effects of uranium ore bodies (7, 10, 11).

6. The environment must be protected (2, 5-8, 12, 13).

7. A careful step-by-step approach should be used for the first HLW disposal system (1, 2, 3).

8. Residual uncertainties*** must be provided for in the program (3).

9. Waste disposal systems should not be dependent on the long-term stability of social or government institutions (3).

10. The program must be flexible relative to potential changes in the nuclear industry (1, 3).

11. The burden of waste disposal should not be deferred to future generations (1).

---

*Isolation means segregating wastes from the accessible environment (biosphere) to the extent required to meet applicable radiological performance objectives.

**Containment means confining the radioactive wastes within prescribed boundaries, e.g., within a waste package.

***Residual uncertainties are those inherent uncertainties in data, modeling, and assumed future conditions which cannot be eliminated.

These 11 pertinent points represent a broad spectrum of viewpoints, including the Natural Resources Defense Council and the National Academy of Sciences. The diversity of viewpoints is evident, for example, in the case of the Interagency Review Group (2, 3) which was composed of representatives of 14 Federal agencies (2). The Nuclear Regulatory Commission was an active (14) but a nonvoting member whose participation by prior agreement did not constitute any endorsement of the IRG findings and recommendations (15). The Interagency Review Group actively solicited the views of the Congress, public interest groups, and the general public through written review, small group meetings, and public meetings (16). A draft report was issued on 9 October 1978 (17). Fifteen thousand copies of the report were distributed, and over 3,300 written comments were received and considered (2, 18) in preparing the IRG recommendations to the President. The wide distribution and extensive review and comment and the IRG finding that a consensus had emerged on a number of planning objectives (19) increase the value of the IRG recommendations in considering the issue of HLW disposal system objectives.

The 11 points listed above were considered along with pertinent information from the literature in developing the NWTS Program objectives described in II.A.1.3 below.

II.A.1.3 National Waste Terminal Storage Program Performance Objectives

Objective 1. Waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Any loss of containment should be a gradual process which results in very small fractional waste inventory release rates extending over very long release times, i.e., catastrophic losses of containment should not occur.

Basis for Objective 1

Waste containment in any given system should provide maximum separation of waste and accessible environment. Although containment may not be guaranteed throughout the entire period of waste toxicity, ensuring containment throughout the period when fission product decay is dominant adds redundancy to isolation measures and adds a further measure of assurance that unacceptable releases to the accessible environment will not occur.

As indicated in Table II.A-1, the thermal output and radioactivity content of a representative spent fuel assembly decrease rapidly during the first few hundred years while fission products decay and relatively slowly thereafter when the decay of long-lived radionuclides predominates.

Table II-1. Thermal Levels and Radioactivity Content for a Spent-Fuel Assembly From a Pressurized Water Reactor (33,000 MWd/MT burnup)

Years out of Reactor	Thermal Output[a] (kW)	Radioactivity Content[b] (curies)
1	4.8[a]	2.5×10^6
10	0.55	4.0×10^5
50	0.25	1.0×10^5
100	0.13	5.0×10^4
500	0.045	2.5×10^3
1,000	0.026	1.7×10^3
10,000	0.006	4.5×10^2

[a]Source: (Reference 20) R.A. Kisner, J.R. Marshall, D.W. Turner, J.E. Vath, Nuclear Waste Projections and Source Term Data for 1977, pp. 41-42, Y/OWI/TM-34, Oak Ridge National Laboratory, Oak Ridge, TN, April 1978

[b]Source: (Reference 21) S.N. Storch and B.E. Prince, Assumptions and Ground Rules Used in Nuclear Waste Projections and Source Term Data, ONWI-24, Oak Ridge National Laboratory, Oak Ridge, TN, September 1979

The rapid decrease in thermal and radiation levels for spent fuel is primarily due to the decay of large amounts of two fission products contained therein, Cs-137 and Sr-90, both of which have half-lives of approximately 30 years. These facts have led to a number of recommendations (5, 8, 22) that control mechanisms be at their maximum effectiveness during the period dominated by fission product decay. Objective 1 requires waste containment throughout that period.

The second condition stipulated in Objective 1 is intended to increase the effective mean time for containment* by requiring containment systems to be resistant to catastrophic failures, i.e., any losses of containment would result in low release rates over long periods of time.

Objective 2. Disposal systems should provide reasonable assurance that wastes will be isolated from the accessible environment for a period of at least 10,000 years with no prediction of significant decreases in isolation beyond that time.

In the context of this objective:

(a) Reasonable assurance means that the preponderance of available technical evidence as interpreted by objective experts in the field supports the conclusions drawn.

(b) Wastes will be considered to be isolated if long-term radiological consequences to the public due to the effects of any reasonably foreseeable events or processes are predicted to be within the range of variations experienced with background radiation. Releases with consequences of a few millirem to a few tens of millirem per year will be considered acceptable provided that the ALARA** standard for man-made systems is met.

---

*Average residence time of waste materials within the containment structure.

**ALARA is an acronym for "as low as is reasonably achievable," which is defined by the Nuclear Regulatory Commission (10 CFR 50.34a(a)) to mean "as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations . . . ."

Basis for Objective 2

As noted in the President's message to Congress on 12 February 1980 (1):

> Our primary objective is to isolate existing and future radioactive waste from military and civilian activities from the biosphere and pose no significant threat to public health and safety.

The "Fact Sheet on the President's Program on Radioactive Waste Management" (1) issued by the White House Press Secretary at the time of the President's statement said:

> The technical programs must meet all relevant radiological protection criteria as well as all other applicable regulatory requirements. Although zero release of radionuclides or zero risk from any such release cannot be assured, such risks should fall within pre-established standards and, beyond that, be reduced to the lowest level practicable.

The very long times over which HLW disposal systems must protect the public health and safety give rise to a number of questions regarding: (i) acceptable levels of risk for disposal systems (as noted above, zero risk cannot be assured), (ii) appropriate time frames throughout which high confidence in system operation must be assured, and (iii) the nature of the assurance to be given.

Objective 2 recognizes that predictions of HLW disposal system performance exceed the length of time measured by recorded history, transcend eras of long-term climatic changes on Earth, and, although short with regard to geologic dating, exceed experience with man-made systems of any type.

The period of time over which waste disposal systems must continue to isolate effectively radioactive wastes has been discussed in a number of forums (8, 12, 22). Some estimates of the required periods of isolation have ranged up to several million years. Such estimates are based largely on a "rule-of-thumb" which would require 10 half-lives for all isotopes to decay to negligible levels. Radioisotopes such as I-129 with a half-life of 1.6 x 10^6 years are frequently cited with no consideration given to the small concentrations and low radiotoxicities of these materials, or extraordinary assumptions regarding human life over the millions of years that are involved. Others (7, 10, 11, 23, 24) have suggested that controls should be exercised until the waste toxicity is below the toxicity of naturally occurring chemical species in the Earth's crust. A subset of the latter concept would compare waste with naturally occurring radioactive ore bodies using the radiotoxicity of each as a common denominator.

In selecting a time-frame over which reasonable assurance should be given, the tradeoff factors are:

1. The duration and the significance of potential radiological hazards as a function of time.

2. The validity of predictions as a function of time.

With regard to Item 1, as indicated by Figure II-1, the radiotoxicity of spent fuel decreases approximately four orders of magnitude (i.e., by a factor of 10,000) during the first 10,000 years and less than two orders of magnitude (i.e., by a factor of less than 100) during the next 1,000,000 years. Also, during the first 10,000 years, the radiological risk due to spent fuel could be reduced below the levels of radiological risk associated with naturally occurring ore bodies, i.e., the risk from the spent fuel properly disposed of would not exceed risks encountered from the uranium ore from which the spent fuel originated.

With regard to Item 2, the magnitude of uncertainties associated with long-term predictions increases with time due to the potential effects of long-term climatic changes and the uncertain technological evolutions of mankind in the future. Long-term climatic and anthropogenic factors make predictions of extremely long-term risks (tens of thousands of years) somewhat speculative (22). Therefore, 10,000 years is reasonable for the period which should receive prime emphasis (with the stipulation that, beyond that time, there should be no strong prediction of unacceptable forthcoming degradations in the isolation capability of the disposal system).

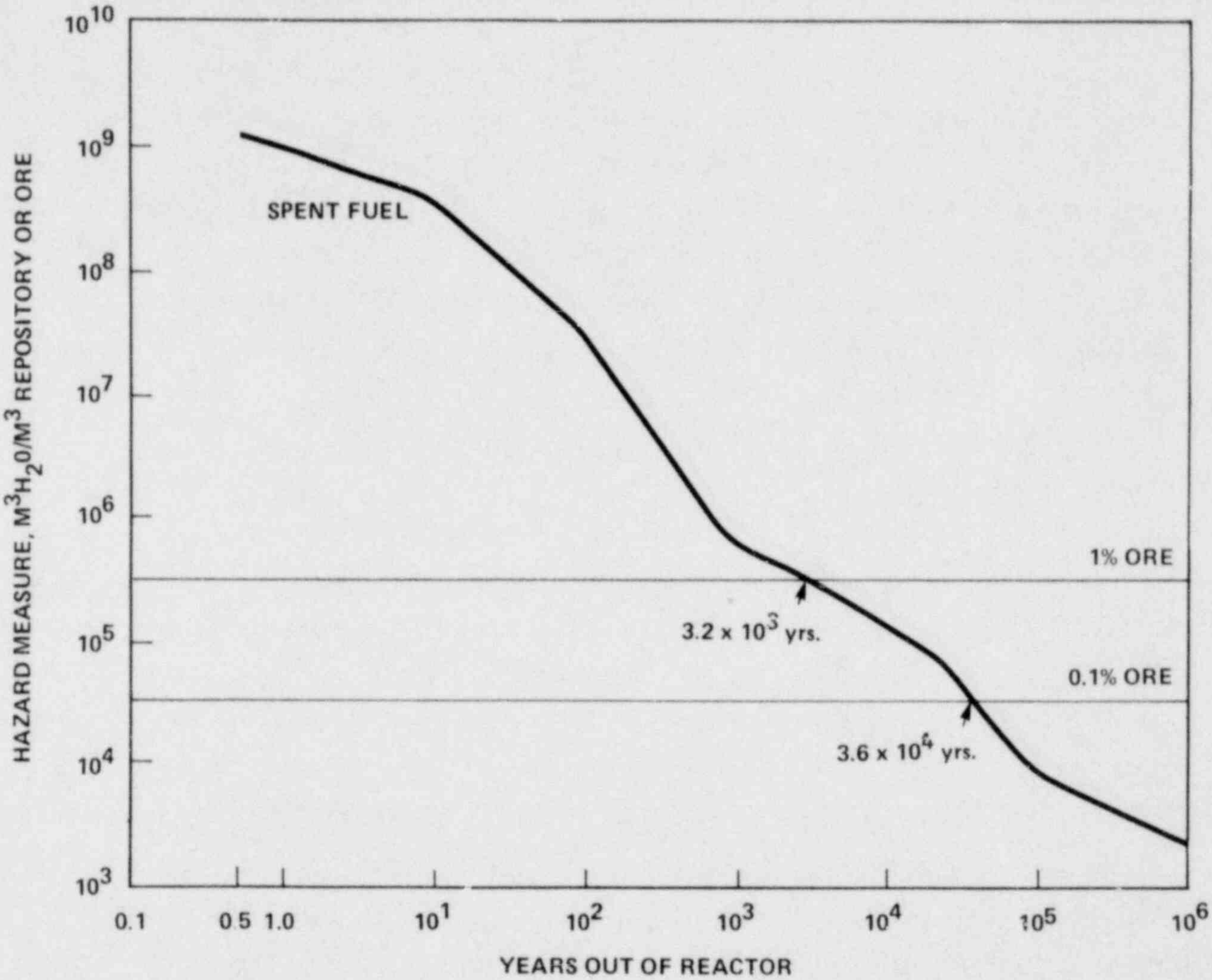


Figure II-1. Hazard Measures Comparison Between Spent Fuel and Uranium Ore

Source: (Reference 25) J.W. Voss, Hazard Measure Comparison of Spent Nuclear Fuel and Uranium Ore, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, April 1980

Part (a) of Objective 2 provides a working definition of "reasonable assurance" which requires a preponderance of technical evidence and objective expert opinions to support conclusions relative to meeting that objective. This definition of reasonable assurance recognizes the existence of residual uncertainties that are inherent in decision processes in any field. The opinion of objective experts is considered to include peer reviews by experts not directly involved in the NWTS ,ram in addition to required regulatory reviews.

With regard to part (b) of Objective 2, the quantification of radiological performance objectives for long-term disposal has been under NRC and EPA consideration for a number of years. The Environmental Protection Agency will issue numerical standards governing long-term releases,* and the Nuclear Regulatory Commission will implement those standards** during licensing.

With regard to establishing a standard by which to measure the significance of releases, natural background radiation has been a common point of reference in nearly all radiological evaluations. For example, the Commission, in its environmental impact statements for reactor licensing, commonly compares doses from postulated routine releases to doses experienced by the same population due to natural background radiation. The relationship between natural background radiation and health effects has been the subject of extensive study. In one study (26), the Commission staff concluded that the information reviewed through the time of that study:

> . . . supports the 1972 BEIR*** estimates that whatever health effects <u>may</u> be caused by natural background radiation, if they exist, they must represent a small part of the total health effects being observed in the real world.

---

*40 CFR 191 (not yet proposed).

**10 CFR 60 (technical portion not yet proposed).

***Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR) of the National Academy of Sciences (NAS-BEIR) Report, 1972.

Although some may protest receiving routine radiation exposures of a few millirems per year from fuel cycle facilities, radiation exposures on that order from other sources are routinely accepted without question. For example, there is no apparent societal discrimination with regard to radiological impacts in choosing between geographic locations in which to live, in choosing between common building materials for housing or in choosing the activities in which to engage. The following are examples of routine radiation exposures:

1. Background radiation variations due to geographic location differences range from approximately 100 to 250 mrem/year within presently populated areas in the United States (8).

2. Nothwithstanding background radiation differences due to geographical locations in 1., above, background radiation exposure to persons living in wooden houses versus brick houses differs by as much as 150 mrem/year (27).

3. Background radiation due to a transcontinental flight in a modern jet airliner is approximately 4 mrem/flight (28).

4. Background radiation from typical domestic activities (e.g., watching TV) is approximately 1.6 mrem/year for an average U.S. citizen (28).

Each of the above-noted activities involves a choice that directly affects an individual's exposure to radiation. The lack of societal discrimination on the basis of the resultant radiological exposure indicates society's implicit acceptance of or lack of interest in low radiological risks compared to the benefits perceived to be associated with these rules. Item 1, above, shows the range of background radiation exposures in the United States to be large. An incremental exposure of a few millirem due to a low probability release from a waste disposal system would be small relative to the variations in background radiation and should be acceptable, since similar or larger variations incurred by human choice are apparently acceptable. The objective suggests that postulated repository-induced exposures should be nearly indistinguishable from background radiation with regard to the magnitude of exposure. For the general population, an incremental exposure equal to a few percent of natural background radiation would appear reasonably low.

In addition to long-term health and safety concerns addressed in Objective 1, an HLW disposal system must protect the health and safety of present generations. Objective 3 is directed toward that purpose.

Objective 3. Risks during the operating phase of waste disposal systems should not be greater than those allowed for other nuclear fuel cycle facilities. Appropriate regulatory requirements established for other fuel cycle facilities of a like nature should be met.

In the context of this objective:

(a) Risks refer to radiological risks either to members of the public or to facility personnel.

(b) Appropriate regulatory requirements refers to safety standards which were derived for similar quantities of radioactive materials and/or systems subject to similar potential modes of failure and which can, with little or no modification, be applied to a HLW disposal facility.

Basis for Objective 3

During the operating phase,* the waste disposal system must function in a manner that adequately protects the public health and safety. As noted by the NRC staff (29), many existing NRC regulations and regulatory guides offer guidance applicable to waste disposal facilities. The quantities of materials handled and the nature of the potential risks associated with the operations of a repository are similar to those presently subject to NRC and EPA requirements (30-35). Such regulations reflect the values accepted by those agencies as appropriate for the public health and safety. Because the Nuclear Regulatory Commission and the Environmental Protection Agency have determined that these levels of safety are acceptable for other nuclear fuel

*The phase when wastes are being received, handled, packaged, and placed into the disposal system.

cycle facilities, it is reasonable to assume that the health and safety of the public also would be adequately protected if the same standards were required during the operating phase of an HLW disposal facility.

In addition to protecting the public health and safety (Objectives 1, 2, and 3), HLW disposal systems also should protect the environment. Objective 4 is directed toward that purpose.

Objective 4. The environmental impacts associated with waste disposal systems should be mitigated to the extent reasonably achievable.

In the context of this objective:

To the extent reasonably achievable means that which is shown to be reasonable considering the costs and benefits associated with potential mitigative measures and reasonable alternative courses of action in accordance with requirements set forth by the National Environmental Policy Act of 1969 and the Council on Environmental Quality.

Basis for Objective 4

As noted by the Interagency Review Group (2), the Environmental Protection Agency (5, 12), the Nuclear Regulatory Commission (6), and others (7, 8, 13), there is a consensus that the environment must be protected and reasonable alternatives should be considered. Consideration of such environmental impacts is required by law (36, 37).

Objective 5. The waste disposal system design and the analytical methods used to develop and demonstrate system effectiveness should be sufficiently conservative to compensate for residual design, operational, and long-term predictive uncertainties of potential importance to system effectiveness, and should provide reasonable assurance that regulatory standards will be met.

In the context of this objective:

(a) Conservatism means taking a course of action in design, analysis, or operation which would tend to overestimate adverse consequences, underestimate mitigating factors, or otherwise provide large margins of safety against undesirable outcomes.

(b) Conservative measures might include:

(i) A careful step-wise approach to design and operation.

(ii) Multiple containment and isolation barriers with sufficient independence and residual effectiveness to assure compliance with appropriate radiation standards over the range of credible failures.

(iii) Design and operating margins which safely limit the effects of system uncertainties.

Basis for Objective 5

The IRG stipulated in its recommendations (3):

> The existence of residual technical uncertainties must be recognized and provided for in the program structure.

Because of the relatively long time frames considered in HLW disposal and the current understanding of HLW disposal systems, there are a number of uncertainties surrounding system parameters, model validities, future conditions on Earth, and certain mechanical-chemical-thermal-radiation relationships.

Regardless of the effort devoted to characterizing, developing, and analyzing an HLW disposal system, uncertainties will remain in the models, data, and other factors influencing performance assessments. Those uncertainties that cannot be eliminated are called "residual uncertainties." The existence of residual uncertainties is not unique to waste management but is common to nearly every complex design or decision-making process regardless of the particular field. As a contingency against such uncertainties, it is customary to design safety factors, backup systems, or other measures to compensate for uncertainties.

Objective 5 sets forth the conditions upon which the NWTS approach to meeting the above IRG recommendation and providing an adequate measure of conservatism is based. An expanded discussion of Objective 5 with particular emphasis on parts (a) through (c) is provided in Section II.A.2.

Objective 6. Waste disposal systems selected for implementation should be based upon a level of technology that can be implemented within a reasonable period of time, should not depend upon scientific breakthroughs, should be able to be assessed with current capabilities, and should not require active maintenance or surveillance for unreasonable times into the future.

Basis for Objective 6

In his 12 February 1980 statement, the President set forth the following requirement (1):

> The responsibility for resolving military and civilian waste management problems shall not be deferred to future generations.

In order to provide waste disposal systems within this century, the technology required for such systems should be available within this decade. Therefore, reliance cannot be placed on scientific breakthroughs. Technology which is "reasonably available" must be brought to bear in a manner that will provide high confidence of successful waste containment and isolation. Confidence in the capability of a technology requires that its performance be predictable by currently available techniques. Given that disposal systems presently under development and available in this century can provide that degree of confidence, there would not be adequate incentives to warrant waiting for other systems requiring extensive technology development to be available.

In addition, as emphasized by the Interagency Review Group (2) and others (7, 12, 22) this generation should implement technologies to dispose of nuclear waste without reliance on perpetual surveillance or maintenance activities. The Environmental Protection Agency, in particular, has indicated in its draft criteria for waste management (5) that the Department should not rely on such surveillance or maintenance for a period of more than 100 years after the termination of active disposal operations.

Objective 7. Waste disposal concepts selected for implementation should be independent of the size of the nuclear industry and of the resolution of specific fuel-cycle or reactor-design issues and should be compatible with national policies.

Basis for Objective 7

In his 12 February 1980 statement, the President set forth the following requirement (1):

> This effort [waste disposal] must proceed regardless of future developments within the nuclear industry -- its future size, and resolution of specific fuel cycle and design issues.

Objective 7 is directed toward implementing the President's directives and ensuring that the proper care is exercised in developing waste disposal system concepts to consider the constraints placed on their use by national policy, to consider their availability for use relative to national priorities, and to consider their overall flexibility to accommodate potential changes in such policies.

Summary

The proposed general performance objectives for the safe and environmentally acceptable disposal of high-level radioactive waste are as follows:

Objective 1. Waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Any loss of containment should be a gradual process which results in very small fractional waste inventory release rates extending over very long release times, i.e., catastrophic losses of containment should not occur.

Objective 2. Disposal systems should provide reasonable assurance that wastes will be isolated from the accessible environment for a period of at least 10,000 years with no prediction of significant decreases in isolation beyond that time.

In the context of this objective:

(a) Reasonable assurance means that the preponderance of available technical evidence as interpreted by objective experts in the field supports the conclusions drawn.

(b) Wastes will be considered to be isolated if long-term radiological consequences to the public due to the effects of any reasonably foreseeable events or processes are predicted to be within the range of variations experienced with background radiation. Releases with consequences of a few millirem to a few tens of millirem per year would be considered acceptable provided that the ALARA standard for man-made systems is met.

Objective 3. Risks during the operating phase of waste disposal systems should not be greater than those allowed for other nuclear fuel cycle facilities. Appropriate regulatory requirements established for other fuel cycle facilities of a like nature should be met.

In the context of this objective:

(a) Operational phase risks refer to radiological risks either to members of the public or to facility personnel.

(b) Appropriate regulatory requirements refers to safety standards which were derived for similar quantities of radioactive materials and/or systems subject to similar potential modes of failure and which can, with little or no modification, be applied to an HLW disposal facility.

Objective 4. The environmental impacts associated with waste disposal systems should be mitigated to the extent reasonably achievable.

In the context of this objective:

To the extent reasonably achievable means that which is shown to be reasonable considering the costs and benefits associated with potential mitigative measures and reasonable alternative courses of action in accordance with requirements set forth by the National Environmental Policy Act of 1969 and the Council on Environmental Quality.

Objective 5. The waste disposal system design and the analytical methods used to develop and demonstrate system effectiveness should be sufficiently conservative to compensate for residual design, operational, and long-term predictive uncertainties of potential importance to system effectiveness, and should provide reasonable assurance that regulatory standards will be met.

In the context of this objective:

(a) Conservatism means taking a course of action in design, analysis, or operation which would tend to overestimate adverse consequences, underestimate mitigating factors, or otherwise provide large margins of safety against undesirable outcomes.

(b) Conservative measures might include:

(i) A careful step-wise approach to design and operation.

(ii) Multiple containment and isolation barriers with sufficient independence and residual effectiveness to assure compliance with appropriate radiation standards over the range of credible failures.

(iii) Design and operating margins which safely limit the effects of system uncertainties.

Objective 6. Waste disposal systems selected for implementation should be based upon a level of technology that can be implemented within a reasonable period of time, not depend upon scientific breakthroughs, should be able to be assessed with current capabilities, and should not require active maintenance or surveillance for unreasonable times into the future.

Objective 7. Waste disposal concepts selected for implementation should be independent of the size of the nuclear industry and of the resolution of specific fuel-cycle or reactor-design issues and should be compatible with national policies.

Section II.A.2 provides an expanded discussion of Objective 5 to more fully describe its meaning and to emphasize its importance in the NWTS program.

II.A.2 The National Waste Terminal Storage Program Conservative Approach to Ensuring Safety

Consistent with Objective 5, the Department has adopted a conservative approach to the safety of HLW disposal systems. The performance objectives in II.A.1, along with the criteria and standards to be promulgated by regulatory authorities, will provide ultimately the basis for assessing the acceptability of specific proposed disposal systems. Extensive application of calculational models will be used to predict the performance of disposal systems under a variety of potential conditions. Inasmuch as the exact conditions that a disposal system will be subjected to during the next several thousand years will not be known with certainty at the time of licensing, it is necessary to build conservatism into the system to compensate for unexpected but credible occurrences which could affect its performance. The IRG report includes the following statement (9):

> Regardless of how minimal hypothesized effects might be, the IRG finds that the Federal Government should maintain a technically conservative approach in pursuing development . . . for high-level and TRU waste disposal.

A "technically conservative approach" means an approach which is seen to be moderate, prudent, and safe. The purpose of a conservative approach is to compensate for perceived uncertainties in the capability to predict events and natural phenomena over the very long periods of time during which radioactive wastes will continue to emit radiation, to compensate for uncertainties in the data, and to compensate for uncertainties inherent in simulating the real world through approximations in modeling processes. An example of the relationship of the application of technical conservatism to the uncertainties perceived to exist relative to using mined geologic disposal is contained in Appendix A of a subgroup report written for the Interagency Review Group (3).

In this section, the conservative measures being incorporated into the NWTS Program to provide confidence in system performance are discussed. For the most part, these measures are not new but have been tried and tested in nuclear and nonnuclear engineering ventures for many years. The main concepts discussed are (i) a step-wise approach, (ii) multibarrier system approach, and (iii) use of design and operating margins.

II.A.2.1 Step-Wise Approach

Whatever the disposal method chosen, the application of that method to high-level waste disposal will be novel. Much of the knowledge brought to bear to ensure the adequacy of the final fully licensed system will be acquired through early testing and observation. It is therefore prudent to continuously reassess the state of knowledge, to search for the implications of newly acquired findings, and to reevaluate designs and plans to ensure that they are supported by the best and latest data. This approach of proceeding cautiously and constantly reevaluating plans in light of new information has been termed a "step-wise approach."

Applied to waste disposal, a step-wise approach might involve initial storage of material in limited quantities under conditions that are well understood or relatively benign. As observed phenomena are better understood, additional quantities of material may be stored under conditions closer to those anticipated for full-scale operation and again all system responses would be checked and understood. The final system operation would be approached in a series of small steps, each one representing only a small extension in the base of well understood knowledge, so that major surprises or unanticipated events of serious risk would be unlikely. For example, if a waste disposal system were found to be unacceptable for some unanticipated reason during the early stages of disposal, a step-wise approach would require reversibility, i.e., the ability to retrieve the waste.

In applying the step-wise approach, the Department is making widespread use of scientific peer review. During the development of the technology to implement a waste disposal sytem, the evalution of phenomena associated with a disposal system, the analysis of possible failure modes, the interpretation of scientific evidence, and the program planning for further development and observation, pertinent information will be subjected to review and criticism by scientific peers of participants in the program. Such evaluation by disinterested third parties with recognized scientific credentials enhances the quality of the program results and increases the likelihood that the safety assessments of waste disposal systems will be adequately performed. This approach is also required in order to obtain "reasonable assurance" in safety-related conclusions in the proposed Objective 2.

II.A.2.2 Multibarrier System

The performance of a waste disposal system can be discussed in terms of those features that provide for isolation and containment of wastes. In order to meet the primary objective previously described for a waste disposal system, namely, to isolate the wastes from the accessible environment and to pose no significant threat to the public health and safety (Objective 2), waste must be prevented from reaching the human environment in quantities in excess of those permitted by radiation standards.* Routes by which the wastes might reach the human environment are called "pathways." Features of the system which act to either contain or isolate wastes to prevent them from reaching the human environment are called "barriers." The multibarrier concept requires that the success of the system be protected against deficient barrier performance or failure by using a series of relatively independent and diverse barriers that would not be subject to common mode failure. Barrier multiplicity is required both as a hedge against unexpected occurrences or failures and to provide appropriate means for protecting against a wide variety of potentially disruptive events. Acceptable system performance must not be contingent on the performance of any non-independent** barrier combinations.

Also as part of the multibarrier system concept, catastrophic (total) system failure must be extremely unlikely due to barrier diversity and independence. (See, for example, proposed Objective 1 regarding potential losses of containment.) Therefore, for any credible event or combination of events and processes, the system would need to retain sufficient barrier effectiveness to keep releases within acceptable levels based on a conservative

*For example, 40 CFR 191 when it is issued by the EPA.

**Subject to common-mode failures.

tive analysis. This defense-in-depth philosophy, i.e., the philosophy of providing several levels of protection to ensure proper system operation, is a natural result of the conservative approach.

Multiple barriers will provide confidence in the capability of the disposal system to perform as required by making the total failure of the system nearly impossible to achieve using credible means. Some examples of barriers which a waste disposal system might include are the following:

1. Long-term resistance of the waste package to potential adverse environments.

2. Design features to inhibit waste mobility, e.g., those to depress radionuclide solubility.

3. Mechanical features of containment structures which inhibit waste transport, e.g., those providing low permeability for fluid flow.

4. Large separation distance and long tortuous pathways between the wastes and the human environment to allow time for radioactive decay.

5. Natural system processes which chemically retard waste migration, e.g., sorption and formation of precipitates.

Specific multiple barriers will be addressed in subsequent sections.

II.A.2.3 Design and Operating Margins

Conservative design practices and operating restrictions will, in part, compensate for residual uncertainties. For example, although there may not be a basis to achieve system optimization, successful system operation can be assured by "over" designing and by "under" operating within prudent limits. It is a common engineering practice to apply factors of safety to designs to compensate for uncertainties in material properties, unexpected operating stresses, and design errors. In a similar fashion, design safety factors will be used in repository systems to ensure successful operation for any set of conditions permitted by the system uncertainties. Furthermore, the system will be operated within a range that has comfortable safety margins between operating conditions and potentially undesirable conditions.

To this end, limiting conditions will be determined and a technical basis for quantifying the appropriate margins will be developed. Subsequent discussions in this part will address sensitivity and uncertainty analyses being conducted to establish a better perspective of the risk associated with a variation of the input parameters.

By utilizing the conservative philosophies discussed in II.A.2, a high degree of assurance in successful system operation will be possible.

Chapter II.B discusses the various disposal methods under consideration and provides the rationale for choosing an interim planning strategy. Subsequent sections indicate the steps taken to develop that strategy, utilizing the philosophies in this chapter to give high assurance that the objectives in II.A.1 will be met.

II.B ALTERNATIVES AND PREFERRED METHOD FOR DISPOSAL

This chapter presents the alternative nuclear waste disposal concepts that have been proposed. Evaluations of these concepts to arrive at a method adopted as an interim planning strategy are presented, along with supporting rationales. The alternative disposal methods are compared to the proposed disposal system objectives discussed in II.A.

II.B.1 Alternative Disposal Methods

Alternative methods for the disposal of spent nuclear fuel have been examined in many studies. Most recently, the Interagency Review Group (IRG) (2) and the Department in its Draft Environmental Impact Statement on Management of Commercially Generated Radioactive Waste (38) have examined the waste disposal alternatives. Other reports examining the broad range of waste disposal alternatives include studies by Battelle Northwest Laboratory (39, 40) and the Environmental Protection Agency (41). The waste disposal technologies which have been considered within these reports are as follows:

1. Mined geologic disposal.
2. Subseabed disposal.
3. Very deep hole disposal.
4. Rock melting disposal.
5. Island disposal.
6. Ice sheet disposal.
7. Deep well injection disposal.
8. Space disposal.
9. Waste partitioning and transmutation.*
10. Chemical resynthesis.*

---

*These technologies are not waste disposal techniques but are pretreatment options, as discussed later. They are presented in this document for completeness.

In the studies mentioned above, these technologies have been evaluated, examined as to their environmental impacts, and assessed as to their technologic state and applicability to disposal of spent nuclear fuel (6). Based on the analyses, it has been concluded that disposal of nuclear wastes in mined geologic repositories is most suitable for adoption as an interim planning strategy pending completion of appropriate environmental review (1, 43, 44). On 12 February 1980, the President adopted an interim planning strategy focused on the use of mined geologic repositories (1). The rationale supporting mined geologic disposal is presented in II.B.1.1 below. The remaining nine disposal technologies are briefly described in II.B.1.2 through II.B.1.10.

II.B.1.1 Rationale for Mined Geologic Disposal

The Department's current programmatic emphasis is toward the establishment of mined geologic repositories, as an interim planning strategy. This section presents the rationale for the primary emphasis on mined geologic disposal.

There are locations on Earth where changes of a geologic nature take place slowly over time periods of millions of years. The rate of change for geologic systems subject only to such long-term change mechanisms would be so low that they could be assumed to be stable for periods of hundreds of thousands of years. Consequently, it is believed that locations within the Earth's crust whose primary change mechanisms require geologic time periods to occur and which appear to provide negligible hydrologic transport potential are suitable for the long-term isolation of nuclear waste (45, 46). To be viable, a rock mass's previous geologic history would need to indicate probable continued stability for at least the next 10,000 years; it should be relatively isolated from circulating ground water; it must be capable of containing waste without losing its desirable properties; it must be amenable to technical analyses (i.e., within man's near-term ability to model); and it must be technologically feasible to develop a repository within it. To effectively use such a rock mass, man must be able to locate it, enter it, emplace waste in it, and seal it without permanently damaging its basic integrity.

As presently conceived, a mined geologic repository will embody three self-supporting and interrelated components to form a complete system for the long-term isolation of radioactive wastes: a suitable repository design, a qualified site, and an engineered waste package system. The repository design will combine conventional mining technology with designs based on the radioactive waste performance requirements. The Department is comprehensively screening the contiguous United States to identify qualified sites (47). The waste package is being designed to maximize containment. The entire system will be required to meet stringent performance and environmental standards (48).

As discussed earlier many groups have examined waste isolation alternatives. In his message to Congress on 12 February 1980 (1), the President adopted an interim planning strategy focused on the use of mined geologic repositories pending the completion of environmental review under the National Environmental Policy Act (50). Two recent reports with broad assessment bases have been the report of the Interagency Review Group (2) and the Draft Environmental Impact Statement (draft EIS) on the Management of Commercially Generated Radioactive Waste (38). The key elements of these reports relative to mined geologic disposal are summarized below.

II.B.1.1.1 Interagency Review Group on Nuclear Waste Management

The IRG (2) recommended the following interim strategic planning basis for waste disposal (49):

> . . . for the first (disposal) facility only mined repositories would be considered. However, three to five geologic environments possessing a wide variety of emplacement media would be examined before a selection was made. Other technological options (for disposal) would be contenders as soon as they had been shown to be technologically sound and economically feasible.

Following the public comment period of its draft report, the IRG reevaluated its position on waste isolation and stated the following (50):

> The IRG has reviewed its judgments about when implementation might be able to begin for the various technology options and still feels that the statements in the draft IRG report are appropriate. The IRG agrees the technical options other than mined repositories might one day become the preferred approach for high level and transuranic waste disposal, but still considers the relative near-term emphasis to be placed on each should be as described within the interim strategic planning basis for high-level waste (described above).

The President has adopted this strategy as an interim planning strategy (1), and the NWTS Program is consistent with it.

II.B.1.1.2 Draft Environmental Impact Statement on Commercially Generated Radioactive Waste

The Draft EIS (38) embodies a comprehensive examination of the 10 disposal alternatives listed at the beginning of II.B.1. The Draft EIS concludes that mined geologic disposal is the preferred option. This option has been adapted as an interim planning strategy, i.e., not final but subject to change until the final EIS is issued. Prior to the issuance of the final EIS, no decisions or commitments that would foreclose alternatives can be made. The Draft EIS findings on mined geologic disposal areas are as follows (51):

1. Media Properties--A mechanically safe repository can be designed in several types of rock.

2. Site Selection--There are no apparent reasons why the site selection and site qualification procedures described (in the EIS) could not proceed despite the present uncertainties in predicting the long-term geologic stability.

3. Adequacy of Data Base--Further research is required to resolve some deficiencies in the data base before repository performance can be confidently predicted.

4. Waste Form and Container Design--The geologic disposal system will be designed on an integrated basis to meet the required level of isolation. The numerous design variables available (for waste form and container design)--protracted cooling, waste dilution, alternative waste forms, canister and overpack design, and repository design--assure that levels of isolation required to ensure public safety can be met.

5. Assessment of Data Base and Analytical Methods--The geologic data base is strong for some media and being developed for others. Short-term geohydrologic assessment methods exist, while longterm predictive geology methods require much work. The data base for spent fuel's long-term stability is limited but is under development. Many promising waste packages designs have evolved and are being evaluated. Some information is available on the relation of human institutions and waste management, but more work is required. The risks of mined geologic disposal have been bounded for short-term analyses, and are estimated to be very small in relation to man made and environmental hazards. Long-term risk assessment is being pursued further.

6. Predisposal Systems for Geologic Disposal--The technology for requisite predisposal systems is well in hand.

Based on these and other assessments within the draft EIS, the disposal of radioactive waste in geologic formations can likely be developed and applied with minimal environmental consequences. As is discussed in further detail in III.C.3, comments received by the Department on the draft EIS have not, at this time, resulted in a change in these findings, which are provisional until a final EIS is published.

In II.B.1.2 through II.B.1.10, which immediately follow, the nine alternatives to mined geologic disposal which are considered in the draft EIS are discussed and compared to the objectives in II.A.1.

II.B.1.2 Subseabed Disposal

It has been postulated that high-level waste in suitable containers could be emplaced in relatively thick stable beds of sediments located in deep quiescent and remote regions of the oceans where slow sedimentation has taken place over tens of millions of years and where continued sedimentation and stability are expected over millions of years into the future (52-55). The key question is whether the sediments form a natural barrier for radioactive waste isolation. The sediments are clay formations exhibiting both vertical and horizontal uniformity, high plasticity, low permeability to water flow, and high capability for ion exchange and sorption. The sediments of interest range in depth from a few tens of meters up to a kilometer, with large areas averaging 100 m to 200 m deep. Directly above the sediments is a benthic boundary layer. This sediment/water interface extends upward from the seabed about 100 m, with the water column extending upward to the ocean's surface. The benthic boundary layer and the ocean waters offer an infinite heat sink, and provide a potential pathway for ion migration and for dilution. However, an understanding of deep ocean physical circulation and biological processes is necessary to address the risk of sediment isolation failure, improper emplacement, or accidents during transport to the disposal site. The concept may have the potential to isolate wastes from the environment for long periods of time. It is expected that the natural barriers could be augmented by an engineered container that could provide containment until the major part of the heat-generating waste had decayed to low levels.

The regions of such beds of sedimentation (called mid-plate, mid-gyre regions) lie away from the seismic and volcanic edges of tectonic plates under the axes of large circulating masses of water called gyres. At these locations and depths, there is little biological activity. Also, because the sediments are primarily a collection of the wind-blown fine dusts from the continents and other fine particulates that have filtered down through the ocean, there is little resource value in these regions. Manganese nodules may be found, but such nodules are common to large areas of the deep ocean bottom (55).and those under the mid-gyre region appear to be low in iron and copper (55).

Several major uncertainties remain for the subseabed option. It has been suggested that the implementation of subseabed disposal of spent nuclear fuel would be in violation of the U.S. Marine Protection, Research, and Sanctuaries Act of 1972 (57) and would therefore require specific U.S. Congressional action before adoption. Similarly, international laws may be interpeted as restricting the subseabed option (58). It is possible, then, that new international treaties would be required prior to adoption of this option. These two policy concerns raise issues as to whether subseabed disposal would meet the proposed Objective 7 (see II.A). Among the environmental questions which need to be resolved for the option are the confirmation that the sediments provide a natural barrier; that migration of radioactive wastes through the sediment and across the benthic boundary layer is slower than the natural radioactive decay process; that the biological and physical ocean processes are understood, to permit assessment of potential release pathways of radionuclides via the ocean and the food chain; and that potential implanation and transportation accidents are assessed. These concerns introduce uncertainties about the concept's ability to meet proposed Objectives 3, 4 and 5 (see II.A).

Results of research to date support the continuing development of the subseabed option, revealing no reason why it should be abandoned. Thus, the IRG has recommended further exploration to resolve the uncertainties remaining. The Department has implemented this recommendation and is funding a continuing R&D effort on the subseabed option. The total number of uncertainties and issues to be resolved is still significant for this option, but efforts to resolve them are proceeding.

II.B.1.3 Very Deep Hole Disposal

The very deep hole disposal concept (59-62) would require that a deep hole (10,000 to 50,000 ft) be drilled and that packaged spent fuel be stacked within it. This alternative, being a variation of mined geologic disposal, hypothetically permits disposal of radioactive wastes at great depths below the Earth's surface. Conceptually, the very deep hole alternative would be developed and scrutinized similarly to the mined geologic disposal technology. Exhaustive siting and licensing procedures would lead to designation of a site. Surface facilities would be constructed that would resemble those conceived for mined disposal. Mobile drilling equipment would be used to develop the deep drill holes. Following waste emplacement, borehole sealing techniques would be employed to isolate the wastes.

The uncertainties for the very deep hole disposal concept are both technical and environmental. There is limited understanding of the fundamental controlling mechanisms at great depths. At the depths involved, our present ability to confirm the geology is severely limited. Remote sensing techniques are not available for these depths, nor is manned examination possible. Verification of isolation is at best intuitive for the very deep hole concept. These questions indicate that verified compliance with the first six proposed objectives presented in II.A is not now possible because of the high degree of uncertainty. Retrievability appears to be prohibitive with the very deep hole concept, making compliance with proposed Objective 5 doubtful. The technology availability for this concept is also uncertain. The deep drilling capability to the tolerances required is questionable. The emplacement technology is a problem because of currently inadequate cable technology and potential problems in ensuring deep hole alignment. The technologic difficulties and uncertainties with this concept are significant.

Very deep hole disposal is therefore not a prime candidate for waste disposal at this time, but investigations of this option will continue, with the possibility that the necessary technology may become available. Parallel interest in developing resources such as petroleum and geopressurized methane from very great depths provides an incentive for investigation of deep-hole techniques which could, if developed, be applied to waste disposal.

II.B.1.4 Rock Melting Disposal

The rock melting concept (63-68) for geologic disposal of nuclear wastes would call for the direct emplacement of the waste in a deep underground hole or cavity. Radioactive decay heat would cause melting of the surrounding rock, which in turn would dissolve the waste. In time, the waste-rock solution would solidify, trapping the radioactive material in a relatively insoluble matrix deep underground.

The rock melting concept has a large number of technologic and environmental uncertainties associated with it. As with the very deep hole concept, our ability to understand the fundamental geologic and hydrologic mechanisms that exist at reference depths (up to 10,000 ft) is somewhat limited. The use of conventional geologic exploration tools to verify conditions of reference depths is uncertain. Manned inspection is not likely to be feasible. In addition, retrieval of wastes from the process is probably not possible. These and other factors limit confidence in this concept's capability to meet the proposed Objectives 1 through 6 (see II.A).

Additionally, the rock melting concept is not suitable for all wastes in the system. The ability of the concept to dispose of spent fuel is questionable. If a mechanical or chemical preparation step were required, then additional wastes would be generated which would not be suitable for disposal by the rock melting concept, thus mandating that some other technology be simultaneously available. For these reasons, the rock melting concept is not a prime candidate technology for the disposal of spent fuel.

II.B.1.5 Island Disposal

Island disposal (69, 70) is a variation of deep geologic disposal which was conceived to increase the distance between man and wastes. As presently envisioned, island repositories would be mined, have waste inserted, be sealed, and perform in essentially the same manner as mined repositories within continental boundaries. Certain islands in the oceans and along the continental margins have the potential for providing the conditions necessary for an acceptable mined repository waste disposal site. These islands are remote from areas of economic activity and population; many are devoid of known natural resources, since they are usually the remnants of long-extinct volcanoes and are made up primarily of basalt rock formations (ocean islands) or formed from igneous rocks not having any special known resource value (continental margin islands). Many of these islands are purported to have a simple equilibrium hydrological regime wherein a stable lens between the rain-fed surface fresh water and the ocean-fed subterranean salt water is established. At the center of this lens, an overburden of fresh water is hypothesized to displace the more dense salt water. It is possible that at the lens center near the boundary between the salt water and the fresh water, there may be a region in which water flow velocities are small and distance to the ocean water is large. These characteristics may be favorable for restricting releases of radioactive materials to the environment.

The meteorological conditions of many island sites is favorable because of exclusion from areas of advancing ice caps and severe climate changes. However, the effects of severe ocean storms and tsunamis would need to be considered in selecting a particular candidate island and depth of disposal for the waste.

Although many of the islands were at one time the centers of violent tectonic and seismic activity, many of the formerly active centers have remained inactive for millions of years and are now located far from active volcanic and seismic regions. Additionally, island waste disposal complexes would have little or no socioeconomic impact if an uninhabited island could be chosen, although the socioeconomic impacts at ports would require examination. Many such islands are the property of the U.S. Government or private individuals but are outside the jurisdictional boundaries of any of the 50 states. Some islands may be subject to international treaty terms.

The island disposal concept has uncertainties associated with its potential environmental impact. There is a potential for dynamic interaction between the fresh and ocean water lenses in island geology, which may preclude confidence in 10,000-year isolation mechanisms (see proposed Objective 2 in II.A). There are technologic uncertainties with the ocean transport of wastes, which, as in the subseabed concept, would be subject to adverse weather conditions. Several political issues, including international issues, may restrict this option. With these uncertainties, and because the concept does not appear to offer advantages over mined geologic disposal, the island disposal concept is not a prime candidate disposal technology.

II.B.1.6 Ice Sheet Disposal

At and surrounding the Earth's rotational poles, there are large uninhabited and desolate areas covered by ice masses thousands of meters thick and extending somewhat uniformly over the polar regions. Where ice sheets cover continental areas such as Antarctica or Greenland, they remain stable for long periods of time. Due to the extreme cold at the poles, the ice is perennial. At depths greater than a few hundred meters, the ice behaves like a plastic and flows to seal fissures and close cavities. Over long periods of time, regions of ice central to the ice field flow to the perimeter and are broken free by weathering and ocean forces to become icebergs. The transport time for flow of ice from the center of the ice cap to its edge is a function of the Earth's climate and is estimated to vary from tens to hundreds of thousands of years through varying pluvial cycle phases.

Use of ice sheet disposal (71-75) as presently conceived would include the encapsulation and transportation of spent nuclear fuel by sea to a polar disposal site located in a region of stable and uniform ice. Canisters would be placed into a hole a few tens to a hundred meters deep and would be sealed over by water poured in place and allowed to freeze. Heat generated within the canister would melt the ice in a region around the canister, and the melt water and waste container, which are more dense than the ice, would slowly settle. This settling would be likely to proceed to the interface between the ice and the underlying rock. Eventually several thousand feet of solid ice would isolate the waste from the surface. The slow flow of the ice might provide isolation for long periods of time, until the region of ice flowed to the ice sheet perimeter and was broken off.

The ice sheet disposal concept is based on the assumption of long-term stability of the polar ice masses. The concept is burdened with a great number of associated technical, environmental, and legal uncertainties (42). The technology for implementing ice sheet disposal would be extremely difficult with the severe weather and temperature conditions presented. Uncertainties about interaction between emplaced wastes and ice masses raise questions about the concept's ability to contain and isolate wastes for the period contemplated by proposed Objectives 1 and 2 (see II.A). If chosen, the concept would also require new or amended international treaties (42).

Environmentally, ice sheet disposal has been estimated to be unsuitable for nuclear waste disposal. Scientists representing the National Academy of Sciences, the Scientific Committee on Antarctic Research of the International Council of Scientific Unions, and the International Commission on Snow and Ice have concluded that the polar ice masses are not suitable for the disposal of radioactive wastes (76). The principal questions about the ice masses' disposal capability are the uncertainty about the stability of an ice mass for at least a 10,000-year period, and the possibility of wastes being mechanically disintegrated by the movement of the ice mass on the basement rock, leading to escape via unknown pathways. The conclusion on the unsuitability of ice-sheet disposal has been echoed by the United Kingdom's Royal Commission on Environmental Pollution (77). For these reasons, this concept is not currently being pursued.

II.B.1.7 Deep Well Injection Disposal

Geologic formations of sedimentary shales that are hundreds of meters in extent under thousands of meters of protective overburden may be essentially isolated from communication with the ground water and thus are potentially isolated from the environment. In many areas, boundary layers of the shale also form boundaries for a porous formation such as sandstone. This porous formation is thereby isolated from the environment in much the same way that the interior of the shale itself is isolated. In shale and in some of the shale-capped but less porous media, large-area horizontal fractures can be made by introducing liquids under pressure until the rock layers physically separate. This process, called hydrofracturing, has been used to stimulate the production of petroleum from well fields.

The deep well injection disposal concept (78, 79) would take advantage of the isolation afforded by these geologic formations, as well as the hydrofracturing and well stimulation technologies commonly used in the recovery of oil. As presently conceived, spent nuclear fuel would be mechanically or chemically processed to produce a liquid or cement slurry for injection by the deep well injection method. The waste would be injected under pressure into the host geologic formation. If hydrofracturing were the disposal method, the cement slurry would be injected into the shale or hydrofractured porous media at a depth of 300 m to 500 m below the surface. For injection of liquid waste into porous media bounded by shales, depths from 500 m to 5,000 m could be used. Methods for preventing nuclear criticality by dilution of the fissile isotopes or dispersion of neutron absorbers or both would be necessary, because the geometric configuration of the final waste reservoir would be uncertain. Considerations of the chemistry of the host rock would be important to ensuring effective criticality prevention.

Deep well injection is a well-defined technology, having been demonstrated for low-level radioactive liquids in the Soviet Union (80, 81) and for cement/grout slurries in the Unite ates (82-84). As mentioned above, this concept would require mechanical or chemical processing of spent fuel; this process would result in significant quantities of intermediate-level, low-level, and cladding wastes which would also require disposal. The concept is not compatible with some of these other wastes, and so some other disposal concept would be required to support the deep well injection concept.

Many uncertainties exist for the concept, which may affect its ability to meet the first six proposed objectives for waste disposal given in II.A. Included are uncertainties about migration pathways in ground water that could preclude injecting a readily mobile, liquid, high-level waste into deep strata. Containment barriers possible through the use of stabilized solid waste forms and high-integrity containers would not be available using this technique. The deep well injection concept probably precludes retrievability of wastes (proposed Objective 5). In addition, the necessity for processing the spent nuclear fuel may conflict with proposed Objective 7 (consistency with national policy) described in II.A. For these reasons, the deep well injection concept is not a prime disposal candidate.

II.B.1.8 Space Disposal

Several space disposal concepts (85-89) have been considered in recent years. The currently favored concept is injection into a circular solar orbit about halfway between Earth and Venus. Orbital calculations indicate that for at least a million years, and probably longer, this orbit is stable with respect to Earth and Venus and would not intersect the orbit of either planet.

Several techniques could be used to place the nuclear waste into orbit. The concept receiving most attention would use the Space Shuttle to lift the waste package with its attached shielding, and an orbital transfer vehicle with a small second stage to near-Earth orbit. The waste package would then be docked and assembled with the unmanned orbital transfer vehicle and be propelled into the appropriate solar orbit. The orbital transfer vehicle and the shielding would be recovered and returned to Earth for reuse.

Space disposal presents several technical uncertainties which may limit the concept's ability to meet proposed Objectives 3 (operational phase impacts) and 6 (performance with available methods) as discussed in II.A. The operational phase uncertainty involves launch failure and the potential inability of the system to retain wastes in such a failure. Vehicle loss probabilities have been estimated to be as high as 0.06/launch (90). Although not every launch failure might release radionuclides, the Royal Commission on Environmental Pollution nevertheless concluded (77), ". . . the consequences of even one failure that resulted in the release of the wastes into the atmosphere would be so serious as to make the method (space disposal) quite unacceptable at present." In addition to these technical uncertainties, space disposal is less feasible for spent fuel than for selected partitioned waste streams, primarily because of launch energy requirements. Specially tailored waste forms may also be required to survive the postulated conditions of accidental reentry, so that the concept would have difficulty meeting the proposed Objective 7 in II.A (consistency with national policy). Aside from this restriction, partitioning of spent fuel radionuclides would result in other waste quantities which would require disposal via some other technology. The Department is cooperating with the National Aeronautics and Space Administration in continued consideration of this alternative, primarily focusing on the possibility of disposal of defense program high-level wastes, but the significant problems described will require resolution before space disposal becomes a realistic alternative.

II.B.1.9 Waste Partitioning and Transmutation

Waste partitioning and transmutation (91-96) is not a disposal concept, but rather a treatment alternative for nuclear wastes. It is presented here to be compatible with the comparative evaluation referenced. Partitioning involves chemical separation of waste constituents to facilitate an optimum method of management. Transmutation refers to a radiation treatment of wastes by which nuclides with undesirable properties are converted to wastes with more desirable properties (e.g., shorter half-life, lower radiation hazard, lower mobility, etc.). The partitioning and transmutation concepts together commonly imply the separation and "detoxification" by transmutation of selected radionuclides. Conceptually, the principal candidates for partitioning and transmutation are iodine, technetium, and certain actinides, which have very long radioactive half-lives. Transmutation concepts include thermal reactors, fast reactors, fusion reactors, accelerators, and nuclear explosives.

Extensive studies of the partitioning and transmutation process have revealed major difficulties. Principally, there appears to be no risk reduction to the process because of technologic limitations (96-99). Use of the process would require that some disposal concept be used to support it. Recent work has indicated that the process may result in an increased radiation hazard during the short term, with no compensating decrease in long-term hazard. These difficulties and uncertainties appear to limit the concept's ability to meet proposed Objectives 3, 4, and 6 given in II.A.

II.B.1.10 Chemical Resynthesis

The chemical resynthesis concept (100, 101) is not a disposal alternative but would provide a waste form alternative. It is presented here for completeness, since it has been discussed in the comparative evaluations referenced. The chemical resynthesis concept, now only in the early conceptual stage, attempts to achieve thermodynamic equilibrium between waste and host rock. This could conceivably be accomplished by means of the supercalcine ceramics processing and products now under development.

The chemical resynthesis concept is actually just a variation in waste form that may provide specially tailored waste packages of very high integrity. The use of this concept will continue to be examined, as part of the effort in materials development needed to support the Department's ongoing programs in waste management.

II.B.2 Summary

Ten waste management technologies have been discussed above. Mined geologic disposal appears most likely to meet all of the proposed objectives in II.A.1. It is believed that locations within the Earth's crust whose primary change mechanisms require geologic time periods to occur and which appear to provide negligible hydrologic transport potential are suitable for the long-term isolation of nuclear waste.

The alternatives of subseabed disposal and disposal in very deep holes appear more amenable to being assessed with reasonably available methods, but questions remain which must be addressed. They do, however, appear sufficiently promising such that continued examination to assess their potential for later development is warranted. The IRG has recommended that these two options continue to be studied to comprehensively assess their capability as backup options (102).

Many of the remaining alternative concepts, however, currently lack enough definition to be judged by methods reasonably available and therefore fail to meet the proposed objective that waste disposal systems selected should be able to be assessed with current capabilities.

II.C PRINCIPAL FEATURES OF A MINED GEOLOGIC DISPOSAL SYSTEM

Chapter II.B shows that mined geologic disposal is an appropriate interim planning strategy which is likely to meet the performance objectives in II.A.1. This chapter presents a general description of the mined geologic disposal system, system components, and features of the components that contribute to meeting the objectives stated in II.A.1. A description of the characteristics of spent fuel and the environmental conditions potentially encountered by the waste package is also provided. The chapter's primary purpose is to provide the reader with a basic understanding of the system as an introduction to the more detailed discussions presented in Chapters II.D through II.F.

II.C.1 Concept Perspectives

In the mined geologic disposal system, containment and isolation will be achieved by emplacing the packaged waste in a repository hundreds of meters below the ground surface at a site selected for its favorable containment and isolation capabilities. Once radioactive wastes are emplaced, in a properly sited and designed repository, credible means for return to the biosphere are few (see II.D). Dissolution of the waste and transport of radionuclides to the biosphere by circulating ground water are considered the principal means by which radionuclides could be released to the biosphere (103, 104).

In evaluating the system's performance, three periods in time--the operational period, the thermal period, and the post-thermal period--and two regions in space--the far field and the near field--can be considered. The periods in time can be described as follows:

1. Operational period--the time when the repository is open and during which waste can be emplaced or retrieved (see II.F.3).

2. Thermal period--the period after closure of the repository when radioactivity levels and heat production are dominated by fission product decay.

3. Post-thermal period--the time following decay of the short-lived radionuclides, during which the radiological hazard is dominated by the decay of actinides and their daughters (105).

The regions in space are usually referred to as the far field and the near field. Although no precise physical boundary separates them, they may be described as follows:

1. The far field refers to the zone that encompasses the host rock, the adjacent rocks, and the entire region in which far-reaching effects may occur.

2. The near field refers to zones within, or closely adjacent to, the repository structure. In many cases, the near field is further refined into the very near field to describe the zone immediately adjacent to the waste package.

II.C.2 System Description

The mined geologic disposal system will be composed of 3 major subsystems: the natural system associated with the site, the waste package, and the repository. Together they provide multiple independent natural and man-made barriers in accordance with Objective 5 in II.A.1. Descriptions of the natural and man-made systems are provided in greater detail in Chapters II.D and II.E, respectively. The natural geologic and hydrologic features of the repository site, as well as the remoteness of the repository (in terms of depth below the surface and distance from water supplies), provide barriers for containing and isolating nuclear waste from people and their environment (II.D). Engineered barriers incorporated in the waste package and repository system provide containment of the waste, delaying the time and retarding the rate of release of radionuclides into the far-field environment (II.E.1 and 2). Prior to repository closure, engineered barriers in the form of container and waste form will aid in protecting both the repository work force and the general public by containing the waste and limiting the potential for its dispersal if the container is breached. After the repository is closed, protective measures will be provided to reduce the likelihood and the consequences of human intrusion into the repository (II.E.3). A conceptual mined geologic disposal system diagram is shown in Figure II-2. The three repository subsystems are described in the following paragraphs.

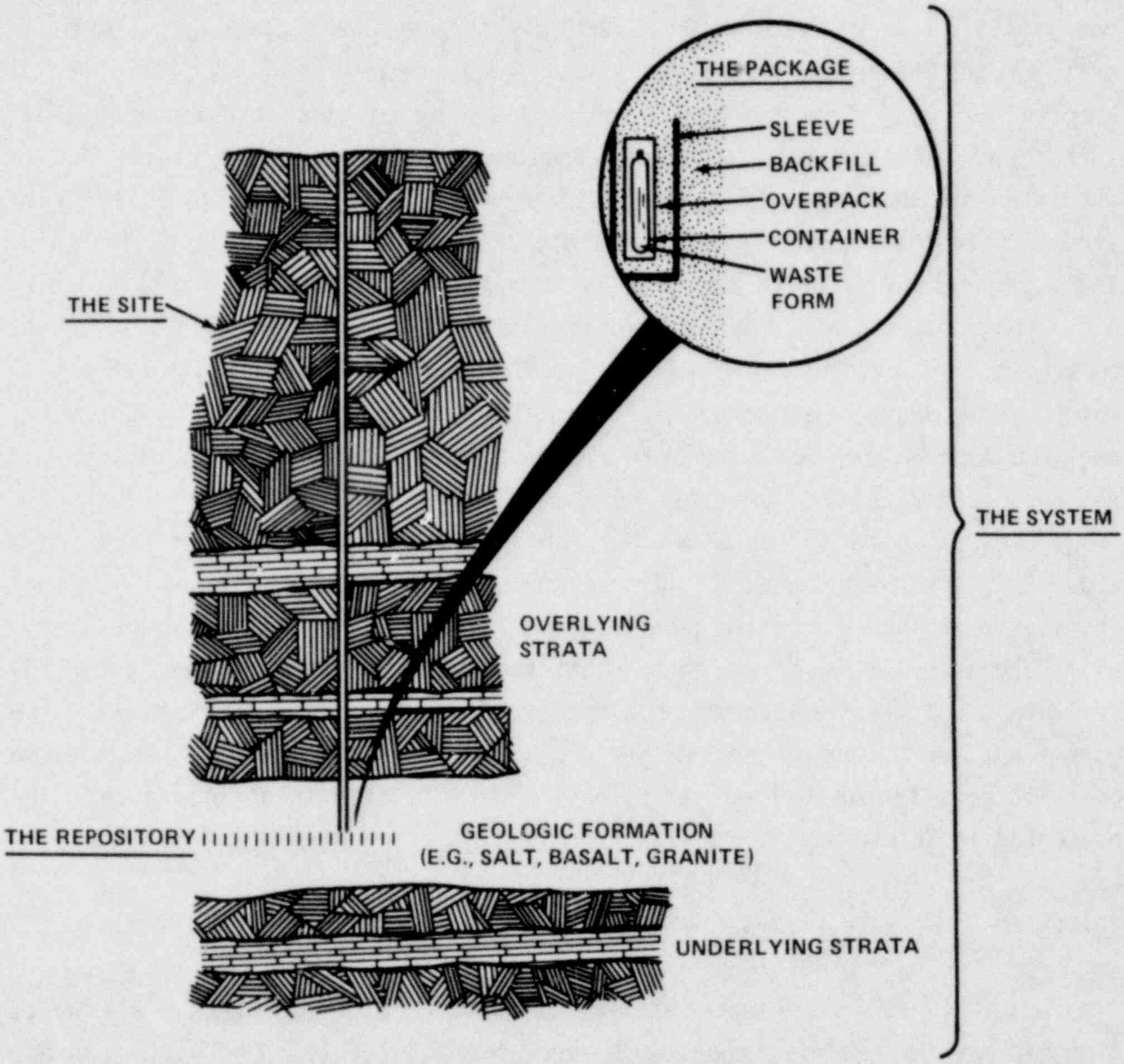


Figure II-2. A Mined Geologic Disposal System

II.C.2.1 Natural Systems

The repository site will include natural barriers which provide waste containment and isolation. These barriers will keep radionuclides from reaching man in unacceptable quantities by (i) maintaining the waste in its emplaced condition for a given period of time (i.e., providing waste containment); (ii) limiting radionuclide mobility through the geohydrologic environment to the biosphere (i.e., providing isolation); and (iii) assisting in keeping man away from the waste (principally by making intrusion difficult, through use of host rock at depth; and unlikely, because there will be few intrusion incentives). The site will contain a host rock suitable for construction of the repository and containment of the waste, as well as surrounding rock formations which can provide adequate isolation. Desirable hydrologic features include low ground water flow rates, long path lengths to the biosphere, and evidence of long-term stability. The important natural attributes of the host rock include its thermal, mechanical, hydraulic, and chemical characteristics, which determine ground water movement and chemistry, and the host rock's ability to withstand thermal effects.

Site selection factors are based on the characteristics cited above and on other concerns such as the protection of the environment and institutional and socioeconomic concerns. Selection of the repository site will take into consideration containment and isolation capabilities; potential present and future environmental impacts, land use conflicts, and resource conflicts; and other potential social, political, and economic impacts on communities affected by the repository. Specific site selection factors are presented in Chapter II.D.

II.C.2.2 Waste Package

The waste package is an important part of the overall waste disposal system. During the operational phase (II.F.3), the waste package provides containment for handling and emplacement and helps ensure retrievability. During the thermal period the waste package provides containment in accordance with Objective 1 (II.A.1). Beyond the thermal period the waste package works in conjunction with the repository system and the na ral systems to provide waste isolation in accordance with Objective 2.

The waste package will include the waste form itself, a stabilizer material, a canister, and one or more layers of protective materials selected to minimize interactions among the waste, host rock, and ground waters. A detailed discussion of the waste package and its components, including the phenomena of importance to its performance, the requirements to guide its development, and the status of knowledge regarding potential materials and components which satisfy those requirements, is provided in II.E.1.

II.C.2.3 Repository System

The repository will incorporate man-made structures, which permit access to the underground facilities and enhance waste containment, and natural barriers, such as the local host rock, to provide containment and isolation after closure. The design, construction, and operation of the repository will be carried out in a manner that preserves the desirable containment and isolation capabilities of the natural system.

Surface facilities will provide for waste receipt, preparation of the waste for emplacement, and transfer of the waste to the underground workings. The surface facilities will be similar to those that have been operated for the handling of radioactive materials over the past several decades and also similar to common industrial mining facilities, for which considerable engineering experience exists. The surface facilities will be required throughout the operational period of the repository.

Repository facilities at depth will include a receiving area for waste packages lowered down the shafts; transfer vehicles to move the waste packages to the emplacement area and into the emplacement holes; and equipment to emplace auxiliary barriers, backfill, and other shielding, as may be required. Underground handling equipment will be operated by repository personnel.

The repository will be constructed using conventional mining techniques. In some of the softer rocks, continuous mining machines using rotary cutter heads would be used. In hard rock repositories, drilling and blasting would be required for shaft and tunnel construction. Waste packages will be emplaced in the series of rooms in holes or trenches cut into the host rock. The volume of rock removed for access and waste emplacement will be considerably less than the volume removed in a conventional mine covering an equivalent area. Conceptual designs have been developed with extraction ratios of approximately 20%, whereas conventional mines reach ratios as high as 90% (see II.E.2).

In the conceptual designs for repositories prepared to date, the emplacement rooms have been approximately 20 ft (6 m) wide by 20 ft (6 m) high and several hundred ft long (106, 107). Emplacement rooms were separated by pillars of undisturbed host rock about 70 ft (21 m) wide. A typical repository with a local thermal power density of 60 kW/acre (15 W/m^2) might occupy a total area of 2,000 acres (810 hectares) and accommodate about 70,000 MTU (about 160,000 assemblies) of spent fuel, depending on the design bases used. Access from the surface would be through several vertical shafts that provide the means to move personnel and materials to the underground area, to remove excavated rock from the underground, and to ventilate the excavated areas. Figure II-3 depicts a typical repository system.

Operational phase monitoring, during and after waste emplacement, will be conducted to ensure that the repository is performing as predicted (see also II.F.3). If a reanalysis of repository performance based on data collected during the operational phase indicates that the repository is not performing as predicted, retrieval of the waste may be necessary. Repository design and operation will provide for waste retrieval capability throughout the operational phase. The repository design will also facilitate the decommissioning of the facility at the end of the operational period to include sealing of the shafts and rooms when authorized by the Commission.

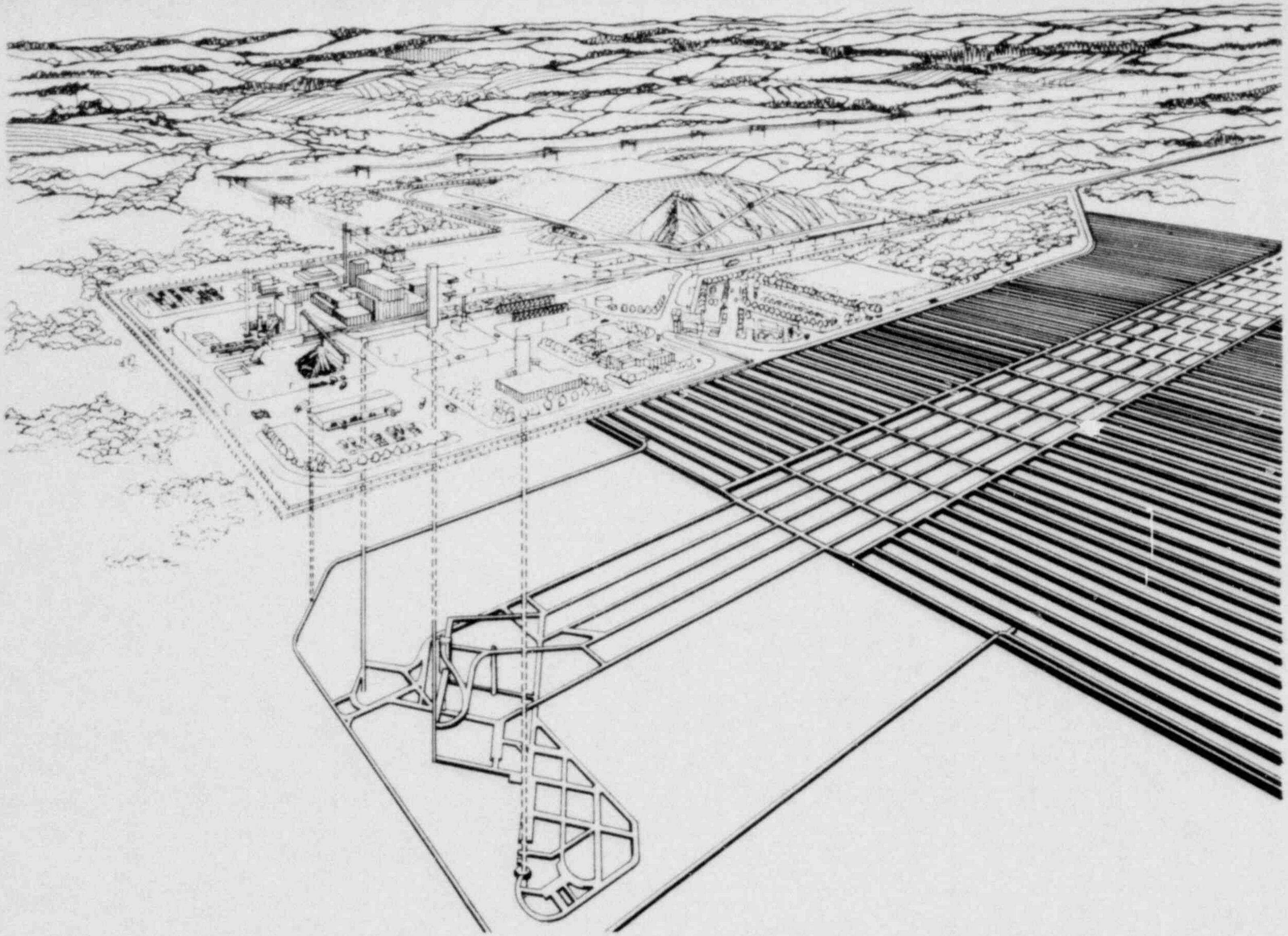


Figure II-3. A Typical Repository

<u>Source</u>: (Reference 107) Kaiser Engineers, <u>Conceptual Design Report - National Waste Terminal Storage Repository for Storing Spent Fuel in a Bedded Salt Formation</u>, Kaiser Engineers, Oakland, CA, January 1979

II.C.3 Spent Fuel Characteristics

As stated in Part I, spent fuel from nuclear power reactors is used in this Statement as the representative waste form. To aid subsequent discussions of the waste package and the impacts of the waste on the man-made and natural systems, a brief summary of spent fuel characteristics is provided here. The reference spent fuel discussed here is uranium dioxide (UO_2) fuel which has typically undergone 33,000 MWd/MTU burnup in a light water reactor (LWR). Light water reactors are either pressurized water reactors (PWR's) or boiling water reactors (BWR's).

II.C.3.1 Physical Characteristics of Fuel Rod and Assembly

Fuel rods for light water nuclear power reactors consist of short cylinders (pellets) of sintered uranium dioxide fuel which are stacked and hermetically sealed in zirconium alloy or stainless steel cladding tubes. The UO_2 (fuel contains slightly enriched uranium in which the fissile U^{235}) content is 2% to 4% of the total U content. Table II-2 summarizes the fuel rod characteristics for LWR's. An example of a typical LWR fuel rod (108) is shown in Figure II-4.

Table II-2. Characteristics of Typical Light Water Reactor Fuel Rods

Characteristic	PWR	BWR
Length, m	3.8	4.1
Active fuel height, m	3.7	3.8
Outer diameter, cm	0.95	1.2
Uranium content, kg	2	3
Pellet length, cm	1.5	1.5

Source: (Reference 109) U.S. Department of Energy, Analytical Methodology and Facility Description - Spent Fuel Policy, DOE/ET-0054, p. I-4, August 1978

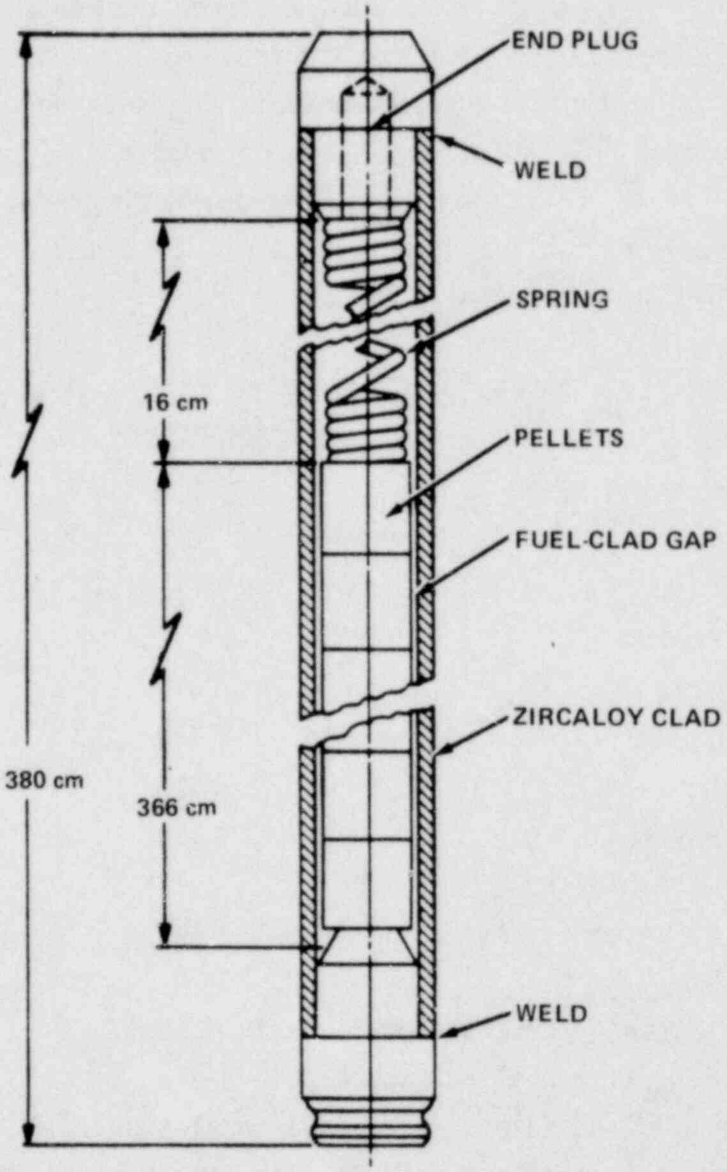


Figure II-4. Typical Nuclear Fuel Rod

Source: (Reference 108) W.H. Baker and F.D. King, Technical Data Summary, Spent Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant, DPSTD-AFCT-77-7, Savannah River Laboratory, p.A.3, E.I. du Pont de Nemours and Co., Aiken, SC, August 1977

The fuel rods are assembled into a square array, spaced and supported by grid structures and end pieces. Although similar in design, fuel assemblies used in PWR's and BWR's differ significantly in size and fuel content. A typical PWR assembly is a 17 x 17 rod array with 264 fuel rods, with the other 25 spaces left open for control rods, burnable poison rods, and instrumentation. A typical BWR assembly is a 7 x 7 rod array. The entire BWR assembly is encased in a thin Zircaloy box called a fuel channel.

Table II-3 summarizes the physical characteristics of typical PWR and BWR fuel assemblies (109). Figures II-5 and II-6 show examples of typical PWR and BWR assemblies (108).

Table II-3. Characteristics of Typical Light Water Reactor Fuel Assemblies

Characteristic	PWR	BWR
Length, m	4.1	4.5
Cross section, cm	21.4 x 21.4	13.9 x 13.9
Array, number	17 x 17	7 x 7
Total weight, kg	670	279
U/assembly, kg	460	190
UO_2/assembly, kg	525	215
Zircaloy/assembly, kg	130[a]	57[b]
Hardware/assembly, kg	16[c]	8[d]
Total metal/assembly, kg	145	650

[a]Includes Zircaloy control-rod guide thimbles.

[b]Includes Zircaloy fuel-rod spacers.

[c]Includes 10 kg stainless steel (SS) nozzles and 5.5 kg Inconel-718 grids.

[d]Includes SS tip plates and negligible amount of Inconel springs. (Inconel is the trademark of International Nickel Co.)

Source: (Reference 109) U.S. Department of Energy, Analytical Methodology and Facility Description - Spent Fuel Policy, DOE/ET-0054, p. I-4, August 1978

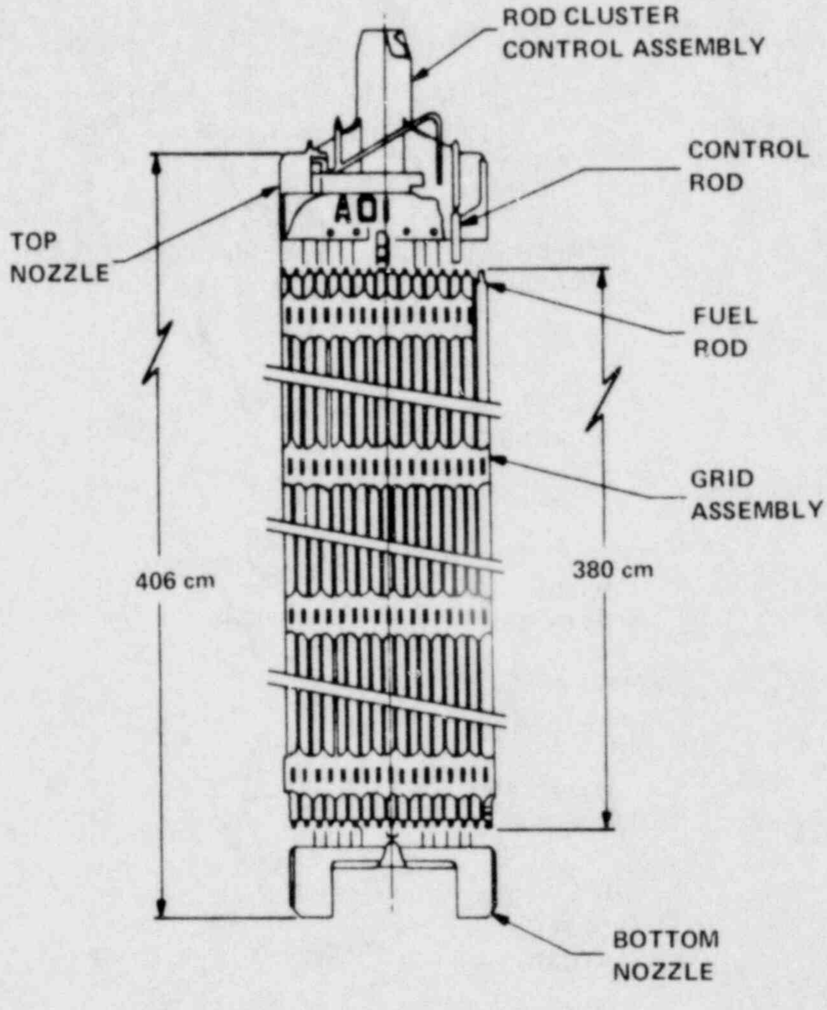


Figure II-5. Pressurized Water Reactor (PWR) Fuel Assembly

Source: (Reference 108) W.H. Baker and F.D. King, Technical Data Summary, Spent Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant, DPSTD-AFCT-77-7, Savannah River Laboratory, p. A.3, E.I. du Pont de Nemours and Co., Aiken, SC, August 1977

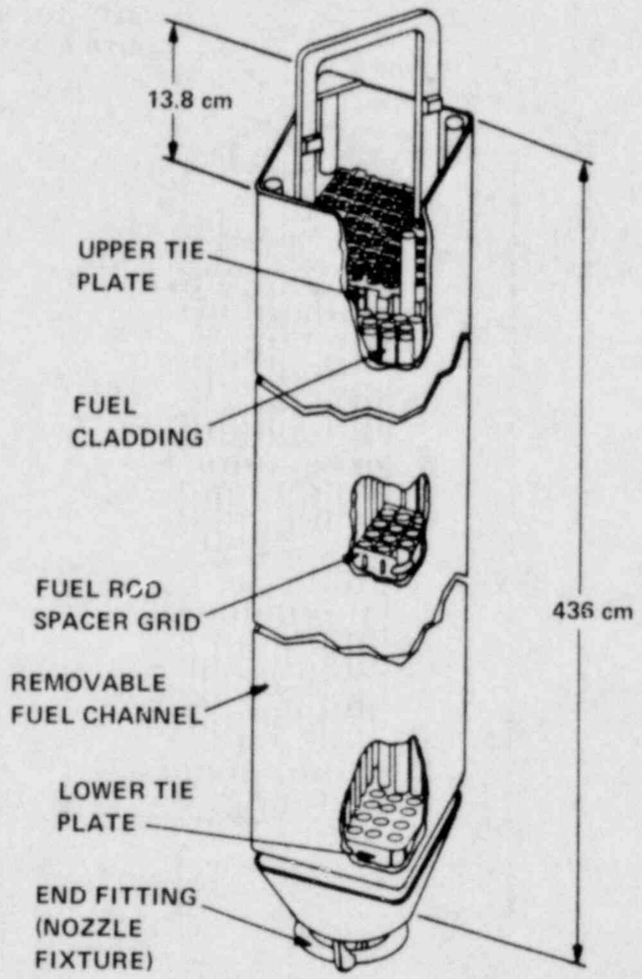


Figure II-6. Boiling Water Reactor (BWR) Fuel Assembly

<u>Source</u>: (Reference 108) W.H. Baker and F.D. King, <u>Technical Data Summary, Spent Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant</u>, DPSTD-AFCT-77-7, Savannah River Laboratory, p. A.3, E.I. du Pont de Nemours and Co., Aiken, SC, August 1977

II.C.3.2 Thermal and Radiation Characteristics of a Fuel Assembly

Most of the radionuclides formed during reactor operation are contained in the structural matrix of the fuel. A small fraction is in the annular gap surrounding the fuel pellets and in the hardware components of the fuel assembly. After an LWR fuel assembly is irradiated, it normally shows no outward external physical change other than the presence of thin films of deposits and corrosion products. However, within the fuel rods, irradiation causes the UO_2 pellets to fracture because of thermal stress and changes in the mechanical properties of the rods.

New fuel can be handled and shipped as a standard commercial product, but spent fuel must be cooled and shielded because it is highly radioactive and produces heat. Initially the heat and radiation in the spent fuel are primarily generated by the radioactive decay of short-lived nuclides. As fuel ages, the radioactivity decreases and the amount of cooling required decreases. The decrease of radioactivity and heat generation for a typical fuel element after removal from a reactor is shown in Table II-4. As a reference case, it is assumed that 10 years will have elapsed after removal of the spent fuel from the reactor before the fuel is introduced into a repository.

The spent-fuel characteristics described above will be referenced in the ensuing discussions of a mined geologic repository. A more detailed discussion of the characteristics of spent fuel is provided in IV.D.4.1.

Table II-4. Thermal and Radiation Characteristics of a Spent Assembly (after 33,000 MWd/MTU burnup)

Age (yr)	Thermal Power[a] (Watts/assembly)	Activity[b] (curies/assembly)	Surface Dose Rate[c] (rem/hr)
1	4,800	2.5×10^6	234,000
5	930	6.0×10^5	46,800
10	550	4.0×10^5	23,400
50	250	1.0×10^5	8,640
100	130	5.0×10^4	2,150[d]
500	45	2.5×10^3	58[d]
1,000	26	1.7×10^3	9.6[d]
5,000	15	6.0×10^2	2.5[d]
10,000	6.4	4.5×10^2	1.8[d]

[a]Source: (Reference 110) R.A. Kisner, J.R. Marshall, D.W. Turner, J.E. Vath, Nuclear Waste Projections and Source Term Data for 1977, pp. 41-42, Y/OWI/TM-34, Oak Ridge National Laboratory, Oak Ridge, TN, April 1978

[b]Source: (Reference 111) S.N. Storch and B.E. Prince, Assumptions and Ground Rules Used in Nuclear Waste Projections and Source Term Data, ONWI-24, Oak Ridge National Laboratory, Oak Ridge, TN, September 1979

[c]Source: (Reference 112) A.G. Croff et al., Calculated, Two-Dimensional Dose Rates from a PWR Fuel Assembly, ONL/TM6754, Oak Ridge National Laboratory, Oak Ridge, TN, March 1979

[d] Derived by applying reduction factors in Reference (113) to values given in Reference (112) at 10 years. (Reference 113: H.C. Claiborne, L.D. Rickertsen, and R.F. Graham, Expected Environments in a Nuclear Waste Spent Fuel Repository in Salt, ORNL/TM-7201 (Draft) Oak Ridge National Laboratory, Oak Ridge, TN, January 1980)

II.C.4 Environmental Conditions in Repositories

In a mined geologic repository the waste package will encounter conditions that are influenced by the rock response to heat and radiation from the waste and by geochemical interactions between the natural environment and the waste. Many factors influence interactions between the host rock and the waste package, the most significant are the temperatures and the composition of ground water to which the package might be exposed. Typical conditions can be established for the local environment around the waste package by using the various calculational models discussed in II.F. Representative examples of typical near-field temperatures and ground water compositions for repositories in salt, granite, and basalt are presented in II.C.4.1 through II.C.4.3. The temperature calculations were performed with the repository characteristics given in Table II-5. In all cases the potential effects of fluids were ignored in performing the thermal calculations.

In these examples the term "thermal power density" describes the amount of waste placed in the repository in terms of thermal power generated by the waste per unit emplacement area of the repository, e.g., kW/acre. The local thermal power density is calculated using the horizontal cross-sectional area of one emplacement room and one room pillar and does not include the areas represented by access drifts or the shaft pillar.

Table II-5. Spent-Fuel Repository Characteristics

Characteristic	Salt		Basalt	Granite
Local thermal loading, kW/acre (W/m^2)	60 (15)	100 (25)	80 (20)	80 (20)
Pitch (along canister row), m	2.7	1.6	1.83	1.83
Distance between canister rows, m	1.7	1.7	2.5	2.5
Canister thermal power, W at emplacement	525		550	550

II.C.4.1 Environmental Conditions in a Salt Repository

Calculations have been performed to assess anticipated temperatures and potential accumulation of fluids in the very near field in a salt repository. Thermal field calculations were performed, and the temperatures and temperature gradients were then used to estimate the maximum influx of brine.

II.C.4.1.1 Temperatures in Salt

Calculations of temperatures in salt have been made using a three-dimensional analysis for a design in which storage rooms at a depth of 2,000 feet in a bedded salt formation contain two rows of spent fuel assemblies in vertical, backfilled holes in the floor (113). In this reference, spent fuel generating 525 W per canister at the time of emplacement were used. Local thermal power densities of 100 kW/acre (25 W/m^2) and 60 kW/acre (15 W/m^2) were considered. The repository characteristics used have been summarized earlier (see Table II-5). Thermal properties of salt were taken from the draft EIS baseline (114).

The temperature histories at the canister midplane, the position of maximum temperature, for these conditions are shown in Figure II-7. The salt temperature peaks at about 95^oC for the 60 kW/acre case, and approximately 140^oC for the 100 kW/acre case. The temperatures of the canister wall peaks at 100^oC and 145^oC for the 60 kW/acre case and 100 kW/acre cases, respectively. These peak temperatures occur within 50 to 60 years after emplacement. These calculations did not include the potential mitigating effects of ventilating emplacement rooms, which, if carried out throughout the emplacement period, could remove a significant fraction of the thermal energy which would otherwise be absorbed by the host rock (113).

II.C.4.1.2 Fluid Conditions in Salt

Under natural conditions, no circulating ground water is present in bedded or domal salt. With the presence of heat, however, the migration of fluids in the form of inclusions, crystal boundary entrapments, and water of hy ation has been suggested as a source of fluid near the waste package (115). This phenomenon has also been observed experimentally (116, 117). Calculations (113) to estimate the influx of brine from the rock salt near the waste canister into the canister cavity have been made with the "MIGRAIN" computer model (see II.E.2 and II.F.1). The temperatures and temperature gradients were calculated in the same manner as those discussed in the preceding paragraph, but for a single-row emplacement configuration at 60 kW/acre thermal power density. MIGRAIN incorporates correlations of laboratory observations of the migration velocity of brine inclusions with the temperature gradient, temperature, and phases present in the inclusion. The laboratory experiments concentrated on single crystal behavior and did not account for migration across crystal boundaries. Vapor phase transport was not considered in the calculation. It is believed that the consideration of these factors would have reduced the calculated accumulation of brine at a waste canister (118, 119). The calculations therefore are considered overestimates of the quantity of brine expected. If the MIGRAIN model is combined with the temperature fields and gradients (maximum value less than $0.3^{o}C/cm$), the inflow shown in Figure II-8 would be predicted. Under these assumptions, for a single-row emplacement configuration, the total accumulated influx of brine 1,500 years after emplacement is about 6 liters.

The concentrations of the chemical constituents of the fluids which could contact the waste package are highly site-dependent. The major constituents are sodium and chlorine. Some magnesium and calcium chlorides and sulfates can also be present. Domal salt generally contains fewer impurities than bedded salt. Thus, brines associated with domal salt have greater sodium and chlorine ion concentrations relative to other dissolved solid concentrations. Brine in inclusions and crystal boundary entrapments may have higher magnesium content. The composition of WIPP "B" brine, which has been used in radionuclide leach tests (120), is shown in Table II-6. This brine was produced by dissolving in water a sample of salt from the Los Medanos bedded salt site in New Mexico; this site has been proposed as the location for the Waste Isolation Pilot Plant (WIPP) project (121).

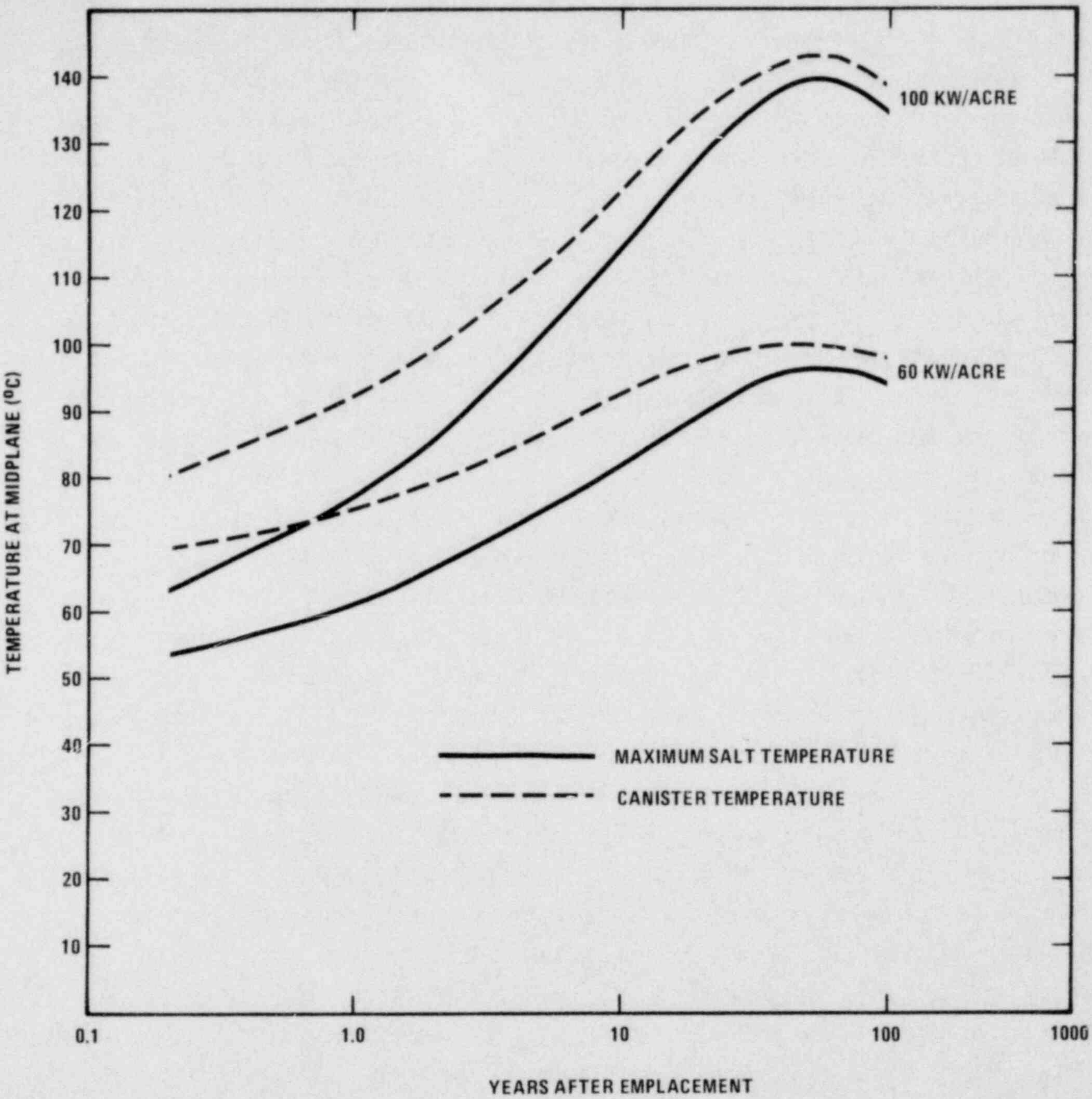


Figure II-7. Temperature Profiles For Salt Repository

Source: (Reference 113) Adapted from H.C. Claiborne and L.D. Rickertsen, Expected Environments in a Nuclear Waste/Spent Fuel Repository in Salt, ORNL/TM-7201, Oak Ridge National Laboratory, Oak Ridge, TN, in draft

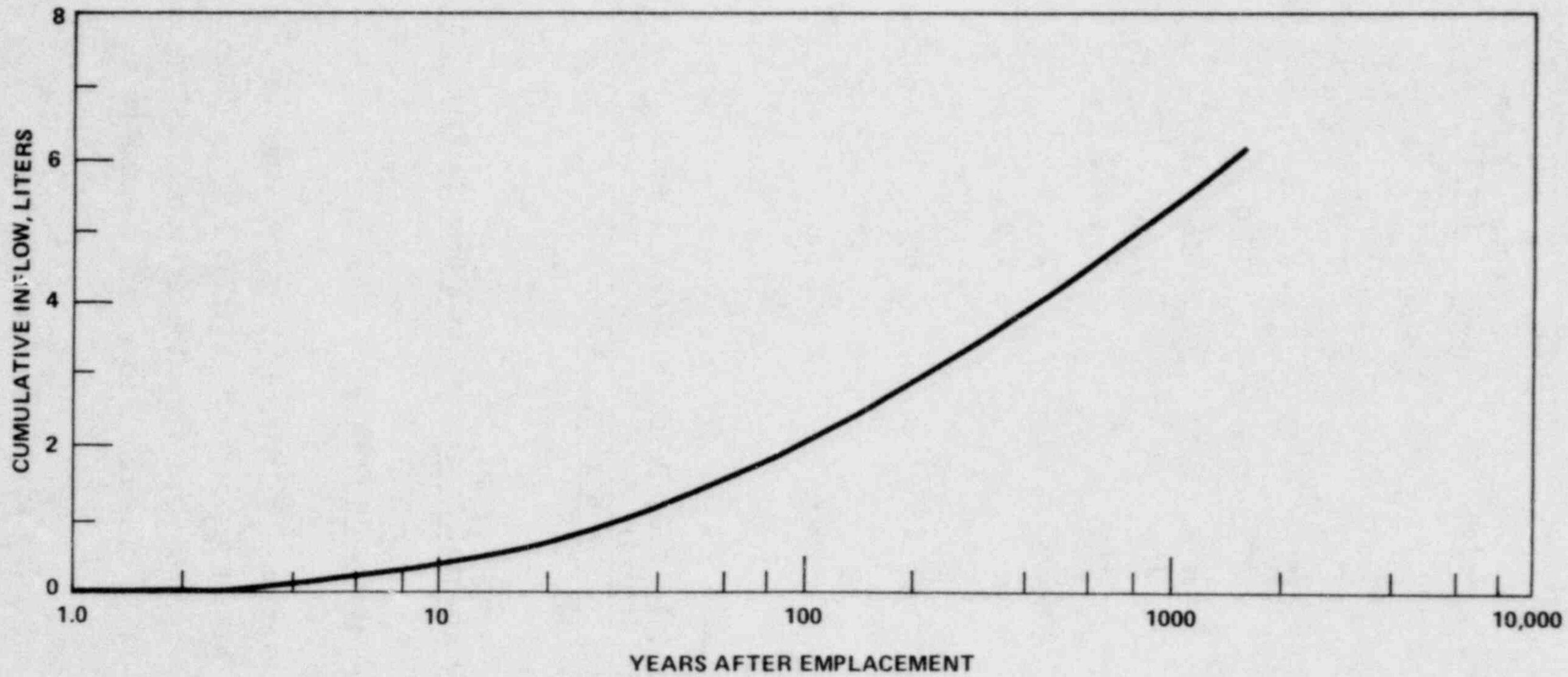


Figure II-8. Cumulative Brine Migration Inflow
(for spent fuel at 60 kW/acre)

Source: (Reference 113) Adapted from H.C. Claiborne and L.D. Rickertsen, Expected Environments in a Nuclear Waste/Spent Fuel Repository in Salt, ORNL/TM-7201, Oak Ridge National Laboratory, Oak Ridge, TN, in draft

Table II-6. Chemical and Ionic Composition of WIPP "B" Brine

Ion	Concentration (mg/l)
Na^+	1.2×10^5
K^+	15
Rb^+	1.0
Cs^+	1.1
Mg^{++}	10.0
Ca^{++}	8.8×10^2
Sr^{++}	15.0
Fe^{+++}	2.0
Cl	1.8×10^5
Br	4.0×10^2
I	10.0
HCO_3	98.0
SO_4	3.5×10^3
B (BO_3)	100.0

Source: (Reference 121) R.G. Dosch and A.W. Lynch, Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico, SAND78-0297, Sandia National Laboratory, Albuquerque, NM, 1978

II.C.4.2 Environmental Conditions in a Granite Repository

II.C.4.2.1 Temperatures in Granite

Calculations of temperatures in the very near field in a granite repository have been performed (122). These calculations were for a reference repository synthesized from the existing designs available throughout the world (123-126). The repository was assumed to be 1,000 m below the

Earth's surface, with spent fuel emplaced in two rows per emplacement room at 80 kW/acre (20 W/m^2) local thermal power density. The emplacement holes were assumed to be backfilled with dry, crushed granite immediately after emplacement of the canister. The reference granite repository characteristics have been summarized in Table II-5. Granite properties were based on a survey of the literature. In this example a thermal conductivity of 2.52 $W/m\text{-}^{o}K$ was used. These properties were assumed to be temperature-independent. Temperature distributions were calculated for a single canister using the finite-element computer model, SPECTROM 41, with an axisymmetric model of the canister. These distributions were combined using SPECTROM 42 to determine the effect of all canisters on the repository temperatures (122).

The temperature histories for granite at the emplacement hole surface and for the canister, both at the canister midplane, are shown in Figure II-9. The granite temperature peaks at 150^{o}C at 35 years after emplacement. The canister temperature peaks at about 170^{o}C at 25 years after emplacement (122).

II.C.4.2.2 Chemical Constituents in Ground Water in Granite

The chemical constituents found in ground water in deep granite are directly related to the composition of the granite containing the water. Since no two granitic rocks are identical in composition, ground water constituents vary (127). However, the major components of the granite that affect the composition of the ground water lie within generally accepted bounds (128) and can thus be used in describing the ground water. Table II-7 lists major and minor constituents typical of an unweathered, intact granite 1,000 m below the Earth's surface, with negligible recharge from the surrounding zones and prior to any interactions with the waste package. These values are based upon a survey of the literature on granitic rock mass properties (127).

Table II-7. Ground Water Composition of Generic Granite

Component	Range (mg/l)
Ca^{2+}	20 - 60
Na^{+}	10 - 100
Mg^{2+}	5 - 30
Fe_{tot}	1 - 20
Fe^{2+}	0.5 - 15
K^{+}	1 - 5
Mn^{2+}	0.1 - 0.5
HCO_3	60 - 400
Cl^{-}	5 - 100
SO_4^{2-}	1 - 40
NO_3^{-}	0.1 - 2
PO_4^{3-}	0.01 - 0.6
F^{-}	0.5 - 3
HS^{-}	< 0.1 - 5
CO_2	0 - 25
SiO_2	5 - 40
NO_4	< 0.01 - 0.1
O_2	< 0.01 - 0.07

Source: (Reference 127) G.K. Coates, Expected Repository Environments in Granite: Ground Water Composition in Granite Rock, Technical Letter Memorandum RSI-0047, RE/SPEC, Rapid City, SD, March 1980

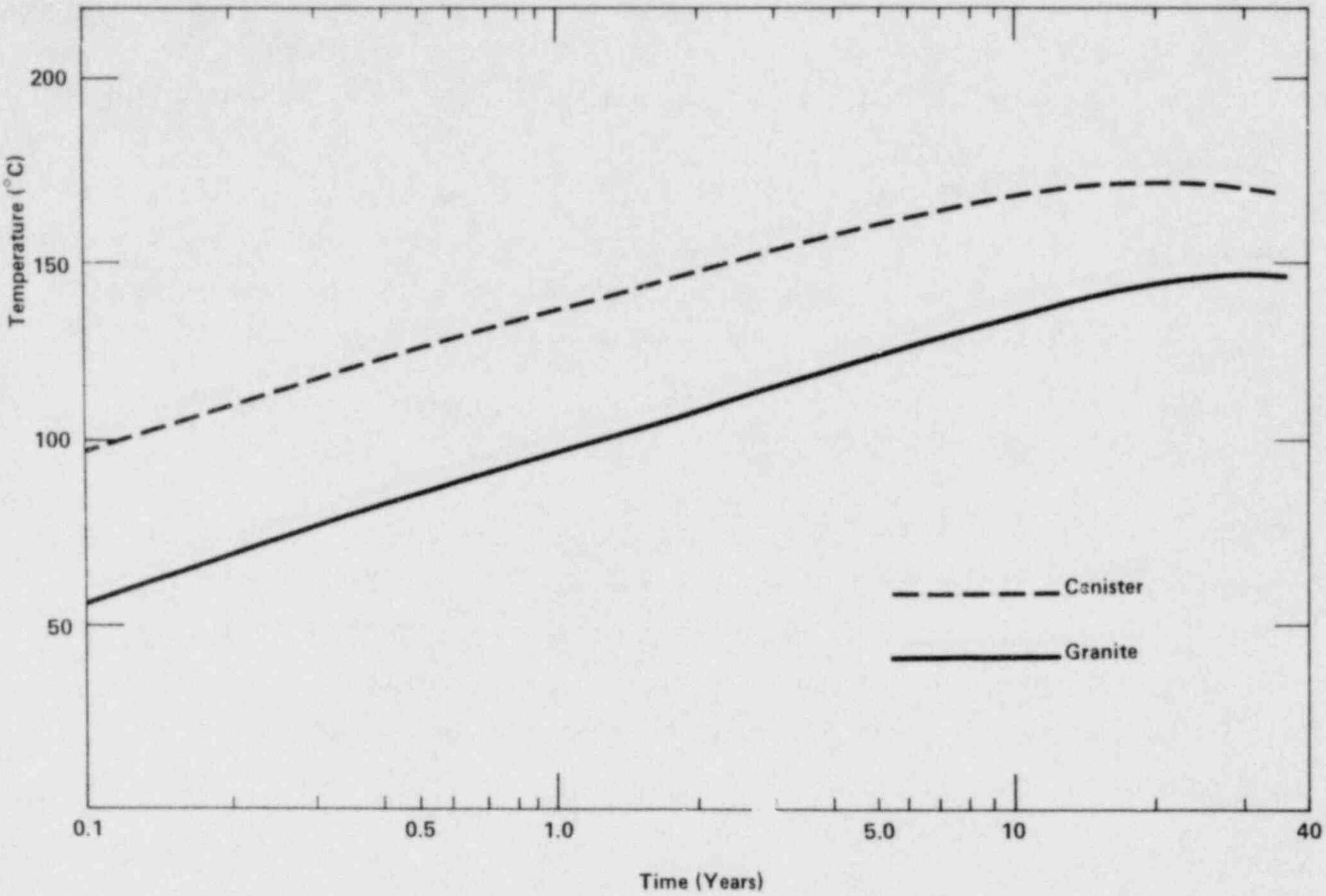


Figure II-9. Temperature Histories in a Granite Repository at Canister Midplane

Source: (Reference 122) J.D. Osnes and K.B. DeJong, Expected Repository Environments in Granite: Thermal Analysis of the Very-Near-Field Region for Spent Fuel Repository in Granite, Technical Letter Memorandum RSI-0048, RE/SPEC, Rapid City, SD, March 1980

II.C.4.3 Environmental Conditions in a Basalt Repository

II.C.4.3.1 Temperatures in Basalt

Calculations for the very near field in a basalt repository have been performed using the same repository and waste characteristics used in the preceding granite example and as given in Table II-5. Typical values for basalt properties were used (129). In this example a thermal conductivity of 1.16 W/m-^{o}K was used. The ambient temperature in basalt at 1,000 m depth is in the range of 40^{o}C to 55^{o}C. A temperature of 50^{o}C was used in this example.

The temperature histories for the basalt at the emplacement hole surface and for the canister, both at the canister midplane, are shown in Figure II-10. The basalt temperature peaks at 250^{o}C at 25 years after emplacement. The canister temperature peaks at 280^{o}C at 20 years (129).

II.C.4.3.2 Chemical Constituents in Ground Water in Basalt

Ground waters within the deeper portions of a thick rock sequence, such as the Columbia River basalts, are cut off from effective interchange with the atmosphere. Hence their chemical compositions are controlled by chemical reactions involving the primary and secondary minerals of the rock. Average compositions and ranges of compositions for ground waters from the Columbia River basalts (130) are given in Table II-8.

II.C.5 Summary

A mined geologic disposal system incorporates those features that can provide waste containment and isolation. The environment which the waste packages will encounter in representative repository systems has been described. It will be composed of three major subsystems: the natural system associated wih the site, the waste package, and the repository. Together they provide multiple independent natural and man-made barriers. The next two chapters describe in detail the features, requirements, and the status of knowledge for each of these subsystems.

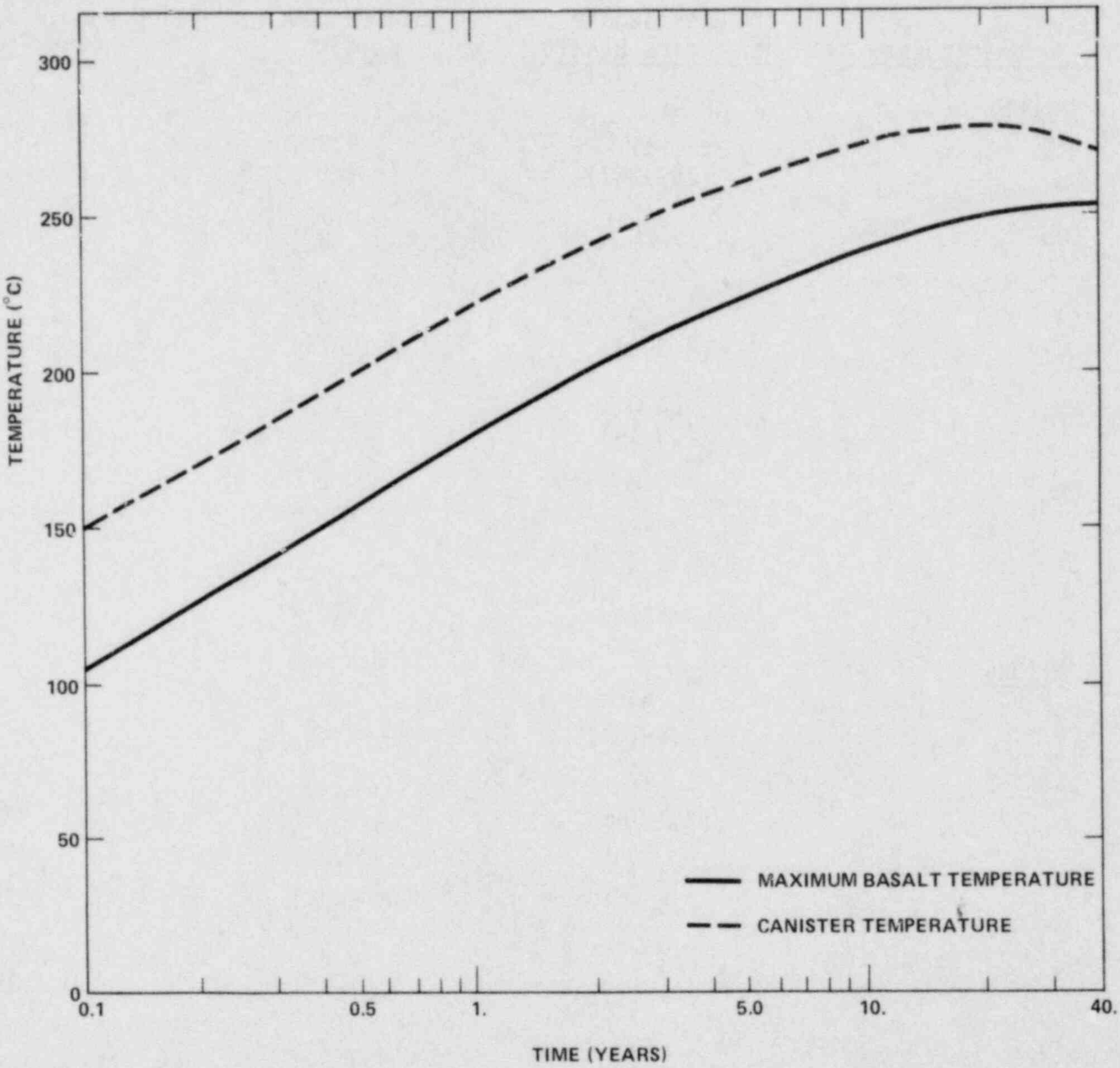


Figure II-10. Thermal Histories for a Basalt Repository at Canister Midplane

---

Source: (Reference 129) Adapted from J.D. Osnes and K.B. DeJong, Thermal Analysis of Very Near Field Region for Spent Fuel Repository in Basalt, RE/SPEC Technical Letter Memorandum RSI-0049, RE/SPEC, Rapid City, SD, March 1980

Table II-8. Average Composition and Range in Concentration of Major Chemical Constituents Within Ground Water for Formation of Columbia River Basalt Group (concentrations in mg/l)

Constituent	Lower Saddle Mountains Basalt	Upper Wanapum Basalt	Grande Ronde Basalt
ANIONS			
HCO_3^-	217 (169-267)	177 (141-216)	75 (66-88)
CO_3^{2-}	0	0	50 (101-127)
Cl^-	20 (4.3-63)	6.6 (3.8-15)	131 (98-148)
SO_4^{2-}	4.0 (.3-18)	11 (.2-32)	72 (13-108)
NO_3^-	.5 (.5)	.5 (.1-2.7)	N.D.[a]
F^-	2.2 (.1-8.0)	.7 (.2-2.0)	29 (22-37)
CATIONS			
Na^+	83 (36-122)	34 (17-80)	225 (182-250)
K^+	11 (7.7-14)	11 (5.9-19)	2.5 (1.9-3.3)
Ca^{2+}	4.7 (.5-22)	17 (1.6-24)	1.1 (.8-1.3)
Mg^{2+}	1.8 (1.-12)	8.8 (.2-15)	.7 (.0-2.0)
SiO_2	69 (56-91)	57 (41-73)	117 (115-121)
Total dissolved solids (sum)	413 (344-505)	324 (283-435)	705 (584-826)

[a]N.D. = not detected.

Source: (Reference 130) R.E. Gephart et al., Hydrologic Studies Within the Columbia Plateau, Washington - An Integration of Current Knowledge, RHO-BWI-ST-5, Rockwell Hanford Operations, Richland, WA, October 1979

II.D NATURAL SYSTEMS OF MINED GEOLOGIC DISPOSAL

This chapter discusses the natural components of a mined geologic disposal system. It describes the function of the natural system (II.D.1), outlines factors influencing that function (II.D.2), lists criteria used by the Department for choosing a site (II.D.3), outlines applicable investigative methods for characterizing the natural system (II.D.4), and summarizes the status of knowledge in current repository exploration programs (II.D.5). Further detailed information regarding current siting activities is given in Appendix B to this Statement.

The natural system described here and the man-made systems discussed in Chapter II.E each provide a high degree of assurance that a repository will perform its function in a manner that is safe and environmentally acceptable (Objectives 2 and 4 in II.A.1). Together, they comprise a total system of multiple independent barriers that will ensure the capability of a mined repository to meet the objectives of long-term containment and isolation.

II.D.1 Function of Natural Systems

For the purposes of this Statement, the natural system is the portion of the Earth's crust, including any fluids within it, that will provide for the containment and isolation of wastes. Natural systems, independent of man-made structures, will provide multiple barriers capable of preventing or retarding the migration of radionuclides from the repository to the biosphere.

The natural system for waste isolation consists of the repository host rock, surrounding geologic formations, and the associated hydrologic environment. It is discussed in the context of a near field and a far field. The near field provides both containment and isolation for the emplaced waste: containment by minimizing the likelihood that circulating ground water will contact the waste package (Objective 1 in II.A.1), and isolation by ensuring that any migration of radionuclides through the near field will be very slow. The prime function of the far field is to ensure that, if radionuclides were released from the near field, ensuing migration to the biosphere would be of sufficient duration to satisfy Objective 2 set forth in Section II.A.1. Understanding the far-field natural system is also important in ensuring the long-term stability of the mined geologic disposal system.

In summary, the natural system must have adequate* containment and isolation capabilities to meet the objectives stated in Section II.A.1. Specific characteristics that determine the capability of the natural system to perform its function at particular sites are discussed in subsequent sections.

II.D.2 Factors Influencing the Choice of Natural Systems

The selection of the natural system in which to site a repository is based on a number of factors related to (i) the function of the natural system itself, (ii) the potential for human breaches of the repository, and (iii) requirements associated with repository construction and operation. For this discussion, the characteristics of natural systems and the phenomena that may affect their performance are grouped into four broad categories: geologic, hydrologic, tectonic, and resource factors (Table II-10). There is some overlap among the categories. For example, hydrologic conditions are closely related to such geologic characteristics as rock types, rock distribution, and the geometric configuration of fractures; also the study of geologic factors is used to determine tectonic processes and the presence of potentially useful minerals. Thus, the separation of the natural system into components is discretionary, but it facilitates study and discussion of the multifaceted geologic environment required for an effective radioactive waste repository.

---

*The terms "adequate," "sufficient," etc., when used with regard to isolation potential, refer to a capability to demonstrate by mathematical models that impacts will not exceed the limits established in regulations to be promulgated by the Nuclear Regulatory Commission and the Environmental Protection Agency.

Table II-9. Factors Which May Affect the Natural System

Geologic Factors	Tectonic Factors
Stratigraphy	Earthquakes
Structure	Fault movements
Sorption characteristics	Epeirogeny, isostacy
Thermal properties	Diapirism
Mechanical properties	Volcanism
	Intrusion
	Erosional incision

Hydrologic Factors	Resource Factors
Hydraulic gradient	Metallic ores
Hydraulic conductivity	Nonmetallic ores
Porosity, permeability	Hydrocarbons
Ground water residence time	Average crustal concentrations
Dissolution	Geothermal sources
Climatic fluctuations	Unique subsurface land uses
Flooding	

II.D.2.1 Geologic Factors

The geologic factors important to the performance of a repository include (i) the stratigraphic distribution and structure of rocks; (ii) the mineralogy of rocks and its relation to the rocks' capacity for radionuclide sorption and (iii) the thermal and mechanical properties relative to natural and repository-induced stress states.

The capacity for sorption and the thermal-mechanical properties of local rock masses are discussed as geologic factors because they are primarily attributes of the rock itself. Hydrologic, tectonic, and resource factors, though closely related to stratigraphic and structural conditions, are discussed later. It is important to note, however, that all components of the natural system depend on the distribution and condition of the various rock types within the vicinity of the site.

II.D.2.1.1 Stratigraphic and Structural Characteristics

A variety of rock types may occur at a particular site, only one of which needs to be at an appropriate depth and of a suitable composition for hosting a repository. Surrounding rock types may also possess attributes that enhance the potential for waste isolation; for example, the clays, siltstones and shales that commonly surround salt domes, layered salt beds, and granitic masses can be highly impermeable to water. The distribution and character of the components of this multimedia rock system, including its structure, determine the containment and isolation potential of the natural setting.

The rock types currently under consideration as host media for a repository are salt, granite, shale, tuff, and basalt. Decades of geologic exploration in the United States have shown that there are many places where potential host rock units of adequate volume exist at appropriate depths (131-133). Several investigations sponsored by the Department and other organizations have confirmed this fact (134-142). Surveys of existing information have identified sedimentary basins where massive salt deposits are present 300 to 1,000 m below the surface (134-137). Similar studies have confirmed the existence of shale and tuff masses with the required dimensions and depth (138, 139). Extensive masses of granitic rocks throughout the United States are also suitable for additional study (140-142). The existence of potentially suitable rock bodies of sufficient dimensions and depth for developing a repository is therefore not an issue. Virtually every State contains a sufficient quantity of one or more of the candidate rock types (131-133). The remaining stratigraphic and structural issues concern the identification of specific sites having suitable combinations of local geologic and hydrologic conditions for waste isolation.

An understanding of the character, condition, and geometric configuration of rocks in the vicinity of a repository is essential for developing predictive models used to estimate the performance of a repository. Promising sites are identified by a screening process that evaluates their geologic settings in terms of certain criteria (II.D.3). Site identification (see III.C.1) is followed by detailed stratigraphic and structural mapping, geophysical surveying, and subsurface exploration (e.g., drilling) which allow the character and configuration of the rocks to be determined in detail. The data thus collected are used to assign numerical parameters to rock properties. The parameters, in turn, are used in computer models to predict the site's containment and isolation qualities. In order to construct models that realistically represent the geologic volumes of interest, it is necessary to determine the degree of correlation between stratigraphic and structural properties and such physical parameters as permeability, sorption capacity, thermal conductivity, and deformation characteristics.

The potential effects of fractures on the thermal and mechanical responses of the rock, on the rock's capacity for radionuclide sorption, and on local ground water flow conditions must be evaluated for each site. If it is found that these effects are sensitive to local variations in fracture geometry or in situ stress, appropriate subsurface characterization and testing methods may need to be developed at each site before final decisions on suitability can be made.

II.D.2.1.2 Sorption Characteristics

Sorption as used in this Statement refers to a variety of chemical processes, including precipitation, ion exchange, surface adsorption, and solid solution formation, that may affix, temporarily or permanently, waste radionuclides onto the matrix of the rock mass within or around a repository. Although the upper limit to the rate of radionuclide transport will be determined by the velocity of ground water movement, sorption by minerals along the paths of ground water flow will retard radionuclide migration, thereby increasing the effective isolation time.

Ground water generally contains low concentrations of the chemical elements that are constituents of the rocks through which the water flows. During flow, a slow interchange occurs between the chemical species dissolved in the water and those in the surrounding rocks. Net deposition or dissolution of specific elements may occur locally, depending on the chemical characteristics of the ground water, the mineral composition of the surrounding rocks, the pH, the oxidation-reduction potentials, temperature, pressure, etc.

The transport of radionuclides by ground water is dependent on a variety of local factors including ground water flow rates; the fracture geometry, porosity, and permeability of the host rock and surrounding geologic formations; the temperature and pressure gradients in the vicinity of a repository; and specific surface area of rocks and fracture-filling materials exposed to circulating ground water. All these factors influence the direction and the rates at which radionuclides can move through the rock in and around a repository. Sorption capacity in this Statement can be expressed by the distribution coefficient K_d, which is the ratio of the solute (particular radionuclide) retained in the rock to the solute remaining in the solvent (water) after equilibrium is attained (147-149). Sorption capacity can also be expressed by the retardation factor R_d, or the ratio of radionuclide migration rates to ground water flow rates (146).

Sorption properties of rocks are characterized by state-of-the-art laboratory and in situ techniques. Studies now in progress are measuring retardation for a variety of radionuclides as a function of temperature, pressure, water flow rate, ground water chemistry, rock type, and mineral composition (143-149). Data from these measurements are used in computer models to evaluate the retardation potential of the host rock and ground water system (see II.F.1). Results indicate that many rock types in diverse settings have a large capacity for retarding radionuclides (143-149).

It is possible to deduce from mass balance relations that the total mass of radionuclides contained in a repository could be sorbed by a volume of rock existing within a few meters of the emplaced cannisters. Migration rates of radionuclides in ground water are thus potentially much slower than the rates of ground water flow. However, the relative rates of sorption and desorption; the surface area of rock in contact with circulating ground water; the effects of fracture geometry, minerals along fractures, temperature and pressure gradients; and the rates and quantitities of local ground water flow are issues that must be addressed for each individual site to estimate its capacity for retarding each radionuclide.

II.D.2.1.3 Thermal and Mechanical Properties

Thermal properties determine the capability of the host rock to absorb and dissipate the thermal energy emitted by the radioactive waste emplaced in the repository. These properties, measured by various techniques (150-151) on rock samples collected at the site during site qualification, are used to evaluate heat transfer and thermal effects in the near field. Parameters that determine the thermal responses of the rock include rock density, the coefficients of thermal expansion, specific heat and thermal conductivity. Heat flow in the vicinity of the repository can alter pre-existing mechanical and chemical equilibria. Stress resulting from thermal expansion in a confined medium will cause deformation which, if it becomes sufficiently great, could cause fracturing or creep behavior in the host rock (152, 153). Knowledge of thermal properties is used in the design of a repository (II.E.2), so that the thermal effects produced by the emplaced wastes will not diminish the containment and isolation capabilities of the natural system. The relationship of thermal properties measured under controlled laboratory conditions to in situ thermal properties of fractured, perhaps water-bearing, rock masses is being addressed by current studies in order to improve the capabilities for thermal modeling at a specific site (II.F.1) (153-159).

Mechanical properties determine the deformation induced in the host rock by both the construction of the repository and the emplacement of heat-generating waste. During site qualification, these properties are measured by various techniques on rock specimens from the site (151-160). The parameters that characterize deformation are Young's modulus, Poisson's ratio, and the coefficient of thermal expansion. Other parameters govern the strength of the rock (resistance to nonelastic deformation); they include cohesion, compressive strength, and tensile strength. Salt and shale exhibit creep deformation and thus have the ability to deform in such a way that they can continually seal potential pathways for ground water flow (152, 161, 162). For these media the creep rate must be determined as a function of stress and temperature. A more rigid medium like granite provides more stable mine openings and makes tunnels and pillars more resistant to failure from thermally induced stresses (153, 163, 164).

Rock failure may not have adverse effects on radionuclide containment and isolation. Specific effects will be determined at each site by comparing the stresses and deformations induced in both fractured and porous media and assessing their impacts on the hydrologic system. The methodology for limiting the impact of repository-induced effects on surrounding rocks by controlling repository design variables are discussed in Section II.E.2.

II.D.2.2 Hydrologic Factors

Knowledge of ground water hydrology is perhaps the most important requirement for understanding the long-term behavior of a mined geologic repository. The transport of radionuclides away from the waste-emplacement zone by moving ground water is by far the most likely mechanism by which radionuclides might migrate from a repository to the biosphere. Understanding the hydrology is fundamental to attempting to predict the directions and rates of radionuclide movement and the locations and concentrations of specific radionuclides at a given time. The pertinent characteristics of ground water systems include the following:

1. Locations and dimensions of water-bearing strata.
2. Existence of aquifers and aquitards.
3. Hydraulic gradient, the driving force for ground water flow.
4. Porosity, permeability*, and transmissivity of the rock mass surrounding a repository.
5. Rates and locations at which the ground water system is discharged and recharged.

---

*Permeability is sometimes used in the discussion as a synonym for hydraulic conductivity, although the dimensional distinction is recognized.

6. Length and direction of potential flow paths (natural or man-induced) from a repository to the biosphere.

7. Travel time from a repository to the biosphere for individual radionuclides.

8. Ground water ages in the vicinity of a repository.

9. Ground water chemistry and its relation to waste rock interactions.

10. Postulated effects of future climates on ground water conditions as deduced from paleoclimatology.

Evaluations of numerous ground water flow systems indicate that hydrologic conditions providing the desired travel times are present in many areas within the United States (165-167) (see also II.D.5). The principal hydrologic issue in repository siting is the confirmation that ground water systems at specific sites are capable of providing the required isolation. Most ground water calculations rely on the assumption that isotropic intergranular flow rather than anisotropic fracture flow is operative (165, 168). The effects of fracture systems on flow conditions at individual sites must therefore be assessed in relation to the containment and isolation capabilities of the site. Many fractures at depth are commonly believed to be tightly closed because of the stress exerted by the overburden (160); therefore natural rock stress is also of interest for hydrologic studies.

Because surface water provides a potential link between ground water and people, the location and characteristics of surface and near-surface water bodies into which the ground water discharges need to be determined at individual sites. Surface water must also be evaluated as a potential source of flooding during repository operation. Available study methods (169, 170) make it possible to identify places that have not flooded for long periods of time and are therefore not deemed likely to be flooded while a repository is in operation.

Potential climatic changes over periods on the order of 10,000 years can alter the hydrologic cycle (171), thus influencing surface water and ground water characteristics (172). The effects of potential climatic changes will be analyzed for specific sites.

In some geologic settings (e.g., evaporite formations), the capacity of ground water to dissolve the host rock must be assessed. Dissolution rates have been measured in some locations considered potentially suitable for a repository. These measured dissolution rates are not sufficient to affect long-term containment or isolation (see Appendix B).

II.D.2.3 Tectonic Factors

Tectonic processes could affect existing natural containment and isolation systems by inducing regional or local deformations in the Earth's crust. Potentially disruptive changes might result from faulting, seismic activity, igneous activity, uplift, subsidence, and alterations in natural stress states. To limit the potential for disruption, repositories will be sited so as to ensure low hazards associated with tectonic activity. Hazards are estimated for each site by analyzing the likelihood and the consequences of a range of tectonic events.

A basic seismic issue in repository siting is the safety of personnel during the operation of the facility. Accordingly, the primary concern is not about long-term radiological effects but about proper mine and surface facility designs that minimize hazards from structural failures should an earthquake occur. Available information suggests that vibratory motion from earthquakes is lower at repository depths than at the Earth's surface (173-176), indicating that surface structures are more likely to be disturbed by earthquakes than the underground repository itself. Many mines currently operate in high seismic risk zones and are even developed directly along potentially active fault zones (177-180)

After the repository has been closed and sealed, the effects of ground motion on the natural containment and isolation system are expected to be minimal. All natural settings are already adjusted to the seismic environment, having attained adjustment over geologic time, during which many earthquakes of all expectable magnitudes for the region have occurred. The potential effects of natural ground motion on the artificial, thermally stressed near field will be mitigated by appropriate conservative design features in the repository relative to expected mechanical behavior regardless of natural ground motion. The effects of fault zones on the hydrologic system are not seismic concerns; they will be addressed for each site as existing structural features of the hydrologic and geologic systems.

The hazards of igneous activity also must be evaluated for potential sites. However, they are not an issue of concern in most of the conterminous United States, even over extended periods of time (181). In areas known to contain centers of Quaternary igneous activity, work is under way to evaluate the hazards associated with its potential recurrence (182-184). Given the vast areas of the country not likely to experience igneous activity, including locally inactive areas in broader, generally active regions, it is feasible and prudent to select repository sites that have extremely low hazards from igneous activity.

Broad uplift and subsidence of the Earth's crust (epeirogeny) are tectonic phenomena that must also be evaluated generally in terms of their potential effects on erosion. Denudation, or the regional lowering of the land's surface by erosion, is not a threat to waste containment or isolation in a deep repository. With the exception of rugged mountainous terrain, maximum denudation rates are only millimeters per century (185), and are incapable of breaching a repository that is 500 to 1,000 meters deep during the time period of concern. The rate of concentrated erosional incision along rivers may approach a rate of a few millimeters per year (180), but such high rates require concurrent tectonic uplift. Lateral erosion can strip hundreds of meters of rock from the surface at rates approaching a centimeter per year (186-187), again only if tectonic processes are active. The potential for vertical incision and lateral erosion will be evaluated for each site to ensure that threatening rates will not be possible at repository locations. In regions where these rates can occur, the distance to a repository from major rivers and steep scarps will be sufficient to preclude a breaching of the system by erosion.

II.D.2.4 Resource Factors

Natural resources occur wherever natural minerals, energy resources, or water-bearing units are sufficiently concentrated for commercial exploitation. Current resource grades are based on demand and the technical feasibility of extraction. In regard to repository siting, there are two concerns about natural resources:

1. The presence and long-term performance requirements of the repository should not preclude the extraction of significant quantities of economically useful mineral deposits or energy resources now or in the future.

2. The concentrations of any minerals or energy resources should be sufficiently low to minimize the chance that future generations might inadvertently breach the repository system through exploratory drilling, mining, etc.

These concerns are addressed by evaluating the geochemical characteristics of potential sites to determine whether unacceptable concentrations of metals, nonmetallic minerals, hydrocarbons, geothermal energy, potable water, or other potential resources e ist in the vicinity of the site. Unique properties of the subsurface that allow the storage of materials other than radioactive wastes may also be considered a natural resource. If inadvertent breaching does occur, the system must be adequate to satisfy Objectives 1 and 2 (II.A).

II.D.3 Requirements for Natural Systems

The preceding discussion has outlined the factors of the natural system that must be considered and integrated during site selection and evaluation. This section presents the requirements for natural systems that are applied during the repository siting program (see III.C).

The following requirements are summarized from the "NWTS Criteria for the Geologic Disposal of Nuclear Wastes" (188), which are presently being circulated for review and comment. These requirements are being used by the Department to guide its research and development activities until such time as formal licensing criteria are adopted by the Nuclear Regulatory Commission and the Environmental Protection Agency. The purpose of the Department's requirements is to ensure that the factors necessary for adequate performance of the natural system are considered and evaluated during the site selection and evaluation process. Other organizations (189-192) have proposed similar criteria that are generally consistent with those listed below. More quantitative criteria will be developed for each study location to guide site-specific decisions on suitability. (See also II.A.)

1. The repository site shall be located in a geologic environment with geometry adequate for repository placement.

    a. The minimum depth of the repository horizon should be such that credible natural processes acting at the surface will not unacceptably affect repository performance.

    b. The thickness of host rock units at the repository horizon should be sufficient to accommodate repository workings and to ensure that impacts induced by repository construction or waste emplacement will not unacceptably affect repository performance.

    c. The lateral extent of host rock units at the repository horizon should be sufficient to accommodate repository workings and to ensure that impacts induced by repository construction or waste emplacement will not unacceptably affect repository performance.

2. The repository site shall have geologic characteristics compatible with waste isolation.

    a. The repository site should have a stratigraphic setting that can be sufficiently defined to permit modeling of the repository system.

    b. The repository site should include host rock units that can be shown to be compatible with the anticipated chemical, thermal, and radiation stresses involved in waste/rock interactions.

    c. The repository site should be so located that the development of disposal areas can be accomplished without undue hazard to repository personnel.

3. The repository site shall have subsurface hydrologic and geochemical characteristics compatible with waste isolation.

   a. The repository site should have a hydrologic and geochemical regime that will prevent radionuclides from leaving the repository and being transported to the biosphere in amounts or levels above the regulatory limits that will be established for the protection of public health and safety.

   b. The repository site should have a hydrologic regime that will allow the construction of repository shafts and the maintenance of shaft liners and seals by state-of-the-art means.

   c. The repository site should be so located that subsurface rock dissolution, occurring or likely to occur, can be shown to have no unacceptable impact on repository performance.

4. The repository site shall be so located that the surficial hydrologic system, both during anticipated climatic cycles and during extreme natural phenomena, will not cause unacceptable adverse impact on repository performance.

   a. The repository shall be so located that nearby surface water bodies, embayments, streams, flood plains, runoff, or drainage under present or future climate conditions can be shown to have no unacceptable impact on repository performance.

5. The repository site shall be so located that credible tectonic events can be shown to cause no unacceptable reduction in repository performance.

   a. The ground motion induced by the maximum credible earthquake should not unacceptably affect repository performance.

   b. The geologic characterization of the site should search for Quaternary faults; if any Quaternary faults are identified, they must be shown to have no unacceptable impact on repository performance.

c. The geologic characterization of the site should search for centers of Quaternary igneous activity; if any such centers are identified, they must be shown to have no unacceptable impact on repository performance.

d. Long-term continuing uplift or subsidence rates must be shown to have no unacceptable impact on repository performance.

6. The repository site shall be so located that likelihood or consequences of past or future human intrusion will cause no unacceptable adverse impact on repository performance.

   a. The repository site shall be so located that future intrusion due to the presence of economically exploitable resources would not cause an unacceptable impact on repository performance.

   b. The repository site shall be so located that the resource exploration history of the site can be defined and can be shown to have no unacceptable impact on repository performance.

The preceding requirements are primarily concerned with the characteristics of the hydrogeologic system. In choosing a site for a repository, there are other concerns about the suitability of the area. These are summarized below:

1. The repository site and its surrounding area shall possess surface characteristics that are compatible with waste disposal.

2. The repository site shall have characteristics that tend to minimize the risk to the population from potential radiation exposure.

3. The repository site shall be located with due consideration to potential environmental impacts, present land-use conflicts, and ambient environmental conditions.

4. The repository shall be sited with consideration to the social, political, and economic impacts on communities affected by the repository.

II.D.4 Investigative Methods

This section reviews the investigative methods for evaluating natural systems in terms of the factors to be considered and requirements for siting discussed in Sections II.D.2 and II.D.3. The status of knowledge obtained by the applied use of the various investigative techniques by current exploratory programs is described in Section II.D.1 (see Chapters II.E and II.F regarding the use of information about natural systems in design and performance analyses, respectively). Table II-10 suggests how the various investigative methods may be integrated into an exploration program; it also indicates the screening phase at which the various methods may be used. The information flow by which the data about the natural system are compiled, interpreted, and evaluated in terms of repository performance is shown in Figure II-11.

II.D.4.1 Geologic Studies

Geologic studies include literature surveys, field mapping, analyses of subsurface data, and laboratory analyses of drill-core and other rock samples. Available maps and reports, as well as remote-sensing imagery and geophysical data for a particular region are reviewed for information on rock types, thicknesses, depths, geometric configurations, rock properties, geologic structures, seismicity, tectonic history, and mineral resources.

General geologic conditions in the United States are well known and have been extensively described (131-133). Geologic data have been collected and analyzed from the time of early expeditions--for example, the 1870's expeditions of the U.S. Geological Survey and early State Geological Surveys (193-198). Exploration for mineral resources--notably oil, gas, coal, and metals--by private industry provides much additional information about sub-surface geologic conditions, in many instances to depths approaching 10,000 m (199). The construction of nuclear reactors, which must meet stringent licensing requirements, has resulted in detailed geologic evaluations of areas in the Eastern, Midwestern, and Far Western United States (200-202). Moreover, various universities have developed as centers of detailed geologic information on specific subjects.

Table II-10. Integration and Phasing of Investigative Methods

FACTORS	METHOD/PHASE[a]														
	GEOPHYSICAL											GEOLOGICAL			
	AIRBORNE						LAND-BASED								
	PHOTOGRAPHY		AEROMAGNETICS	GRAVITY	S.L.A.R.[b]	T.I.R.[c] False Color	MAGNETIC	GRAVITY	SEISMIC	ELECTRIC	DOWN HOLE LOGGING	GEOLOGIC MAPPING LITHOLOGY	SUBSURF. EXPL. STRATIGRAPHY	LAB TESTING RK SAMPLES FOR ENG. PARAMETERS	SUBSURF. TEST FACILITIES
	SPACE	AIR													
GEOLOGIC	NR	RAL	RA	R		RAL	AL	AL	AL	RA	AL	RAL	AL	AL	L
HYDROLOGIC	NRA	RAL				RA			L	AL	AL	RAL	AL	AL	L
TECTONIC	NR	AL	RA	RA	RAL	RAL		AL	AL	L		RAL	AL	L	L
RESOURCE		AL	RA			AL	AL		AL	AL	AL	RAL	AL	AL	

[a]PHASES: N - NATIONAL
R - REGIONAL
A - AREA
L - LOCATION

[b]S.L.A.R. IS "SIDE-LOOKING AIRBORNE RADAR"
[c]T.I.R. IS "THERMAL INFRARED"

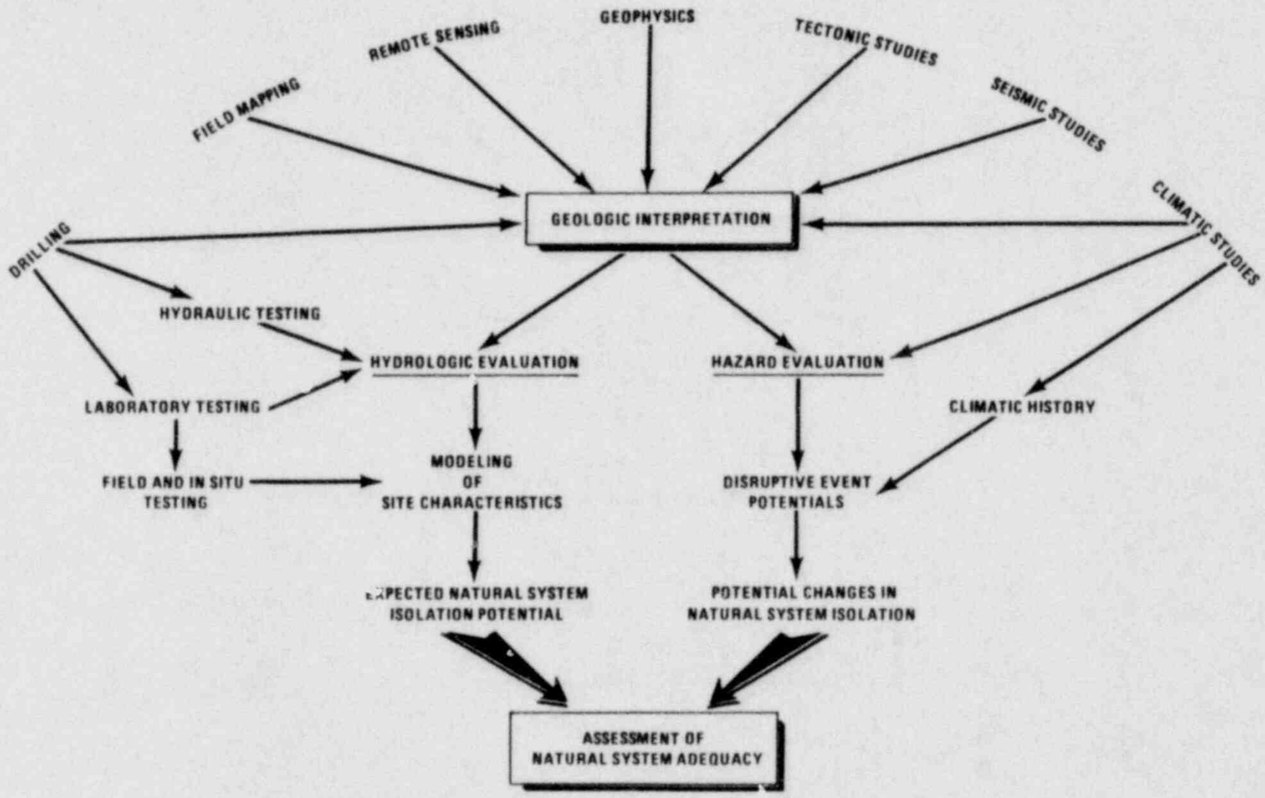


Figure II-11. General Flow of Information About the Natural System

The accumulated knowledge is sufficient to identify areas in the United States that meet many of the requirements (II.D.3) for radioactive-waste repositories. In addition, the previous geological work has resulted in the evolution of a broad suite of investigative techniques that, when selectively used in conjunction, allow highly accurate characterization of the geologic setting of any chosen area. The knowledge that must still be gathered pertains to site-specific factors, and the techniques for gathering it are presently available (203, 204). This section describes some of the various techniques being used by ongoing repository exploration programs (II.D.5) to obtain the needed data.

As exploration efforts focus on a site, field mapping is undertaken to compile or augment available maps of stratigraphic, structural, geomorphic, and tectonic features. Although field mapping is conducted according to highly individual techniques, a standard practice is to record field observations and supporting data on aerial photographs and/or topographic base maps (205). Data provided by remote sensing, geophysical surveys, subsurface borings, and sample analyses are an integral part of the information set on which the geologist draws. In a subjective process, geologic information is analyzed in terms of multiple working hypotheses until it yields a "picture" of the relationships among all the geologic variables (185, 195, 206). Based on the geographic density of the data collection points, reasonable bounds can be established for interpretations of geologic conditions. As more data become available, the "picture" comes into ever sharper focus. Continuous peer review is essential; it serves not only to evaluate conclusions but also, and perhaps more importantly, to guide the geologist responsible for determining the character and the condition of unseen rocks.

The geologic characterization of a site is approached by a series of approximations. Drilling is important for testing previous approximations and supplying additional data for subsequent refined hypotheses. Borehole cuttings and cores allow direct observation of materials from selected sampling points below the surface of the ground. Holes are drilled in locations determined as having the greatest potential for resolving uncertainties. A few appropriately located holes can yield sufficient subsurface data to allow an accurate interpretation of the three-dimensional nature of rock conditions throughout an area on the order of tens of square kilometers. The holes are cored, logged, and tested as appropriate.

Cores are supplied for laboratory analyses of such physical properties as mineral composition, chemical characteristics, porosity, and thermal-mechanical properties. Selected core and field samples are dated by isotopic methods (207, 208) to help determine the history of geologic processes, including fault ruptures, volcanic eruptions, and diapirism.

If required, seismic monitoring nets are deployed around potential sites to provide data for assessing ground-motion hazards. These data may be supplemented by geologic and geomorphic mapping to provide a deterministic basis for the analysis of seismic hazards. The mapping of stream gradients, terraces, hillslope gradients, and drainage patterns also supplies data for estimating uplift, subsidence, the effects of climatic changes, and tectonic history.

Fracture analysis of surface outcrops and drill cores is performed to help estimate the likely density and geometry of fracture systems at the depths of interest. Such systems can only be statistically sampled, and rosette diagrams are commonly used to show the orientation of fractures (160). Histograms of the number of fractures as a function of depth are a means for showing frequency, density, and condition of fractures (209, 210). Because not all fractures are capable of transmitting fluids in significant amounts (211-214), apertures and secondary filling minerals are sampled, analyzed, and perhaps dated to characterize fluid migration capabilities and history.

In general, geologic studies are the mechanism by which all available data about the subsurface environment are synthesized and coordinated to assess whether the stratigraphic and structural settings of a proposed site are suitable for a waste repository. Remote sensing and geophysical studies, discussed next, are conducted primarily to support this activity. Geologic interpretations are the basis for defining models by which the hydrologic, geologic, geochemical, thermal, and mechanical characteristics of a repository are assessed (see Figure II-11).

II.D.4.2 Airborne Remote Sensing

Airborne remote sensing is performed by various techniques, ranging from aerial photography to satellite imagery, that all result in a directly created or a computer-generated image of a particular geographic area of the earth's surface. It is used at all phases of the siting program but is particularly valuable in the early regional studies because it yields large amounts of reconnaissance data in short periods of time (215). Regardless of the size of the region being investigated, remote sensing, in conjunction with field reconnaissance for checking interpretations, is an excellent tool for establishing, compiling, and mapping a data base for the natural setting. Remote sensing imagery for many regions is available and can be readily obtained.

Aerial photography is the most widely used form of remote sensing imagery and is used for compiling topographic maps and for constructing detailed thematic site maps (216, 217). It uniquely reveals many surficial geologic features and is commonly used to identify and map landforms, rock types, unconsolidated surficial deposits and soils, geologic structures, and surface drainage patterns. Specifically, it may indicate the orientation of stratified rocks; the type of surface rocks; contacts between different rock types; the locations of igneous intrusives, salt domes, structural domes, arches, and other folds; the location and orientation of faults, fractures, or joint systems; and karst terrain (e.g., cave systems), breccia pipes, and other surface manifestations of dissolution. Surface water bodies, vegetation assemblages, and man-made features such as petroleum or water well fields, irrigation facilities, and archaeological sites are visible on many aerial photographs.

Color infrared photography is especially useful for differentiating many features related to subtle differences in soils, moisture content, small-scale topography, and rock type (218). Thermal and color infrared photography can indicate the presence of near-surface cavities or linear openings, which influence the amount of moisture at or near the surface (219). Infrared imagery is also used to detect such geothermal features as hot springs which may indicate potential volcanic activity (220). Mineralized zones with economic potential may sometimes be discerned from remote sensing imagery (221).

Computer processing of airborne or satellite spectral data allows rapid classification of vegetation patterns and certain geologic features (222, 223). Analysis of spectral imagery obtained at the same place at different times exploits the fact that annual and diurnal heating and cooling and seasonal changes in vegetation cause temporal differences between rock surfaces which can be resolved by spectral scanners (224, 225).

Where the Earth's surface is largely obscured by persistent cloud cover, side-looking airborne radar imagery (SLAR) can give a reasonable representation of surface features (226).

II.D.4.3 Geophysical Surveys

Geophysical surveys are an integral part of site selection and characterization studies. Many of the geophysical techniques utilized by the petroleum and mineral industries have been applied to the search for geologic repositories. The broad categories of exploration geophysics summarized in this subsection are gravity, magnetic, electrical, and seismic methods. In addition, well logging and borehole geophysics will be discussed.

Geophysical methods can be broadly divided into passive techniques, which measure existing earth fields, and active techniques, which measure the earth's response to an applied stimulus. Gravity, magnetic, and some electrical surveys are passive methods; seismic reflection and electrical resistivity are examples of active techniques. Passive exploration methods are particularly applicable to reconnaissance exploration because they cover large areas fairly rapidly and cost less than active techniques.

Both gravity and magnetic techniques have similar mathematical foundations in potential field theory but measure different physical properties of the earth and use different field methods. Gravity surveys detect small variations in the Earth's gravity field (227). The variations of principal interest to repository siting result from lateral variations in subsurface rock density. These density variations may result from deformed strata, faults, igneous intrusives, diapirs, breccia pipes, or lithologic changes. Topography, elevation, geographic position on the earth, and earth tides also affect gravity values and must be accounted for in determining the portion (anomaly) of the gravity measurement that is due to subsurface effects. The maximum practical precision for field gravity measurements is about 0.01 milligal, or approximately 10^{-8} of the earth's total field. This precision requires that the elevation be determined to an accuracy of about 5 cm at each gravity station. Aerial gravity surveys have a much lower precision (approximately 5 milligals) and are useful only in delineating gross crustal structure (228).

Magnetic methods detect variations in the Earth's magnetic field (227). Magnetic field values vary with the geodetic location and may vary at a given location with time. The magnetic variations (anomalies) of interest to site studies are due to lateral changes in mineral content (especially magnetite) or to variations in the remnant magnetism of igneous rocks. Subsurface structures like anticlines or faults, can be detected if they result in lateral changes of the above properties (229). Measurements can be obtained in terms of the total magnetic intensity, vertical gradients, or oriented (horizontal and vertical) components of total intensity. The practical precision for state-of-the-art aeromagnetic surveys is about 0.1 gauss, or 10^{-6} of the Earth's field. Magnetic surveys are most commonly made from low-flying aircraft, although surveys of limited extent over selected targets of interest can be made at the ground.

Both gravity and magnetics inherently yield nonunique interpretations. However, limits on the allowable interpretations are usually provided by measured rock densities, magnetic susceptibilities, general depth to the causative structures, or seismic profiles from the area of the survey. When sufficient ancillary data are available, interpretations of gravity and magnetic surveys can provide valuable information about regional and local subsurface geology. As a generalization, features can be interpreted if they have dimensions of 1% to 10% of their depth of burial (230).

A variety of electrical methods (227, 231) is used in geophysical exploration; all depend upon detecting variations in the electrical resistivity of the media through which a current flows. Subsurface resistivity is highly variable and strongly influenced by the amount and the nature of fluids in the rocks. For this reason, such hydrologic features as dissolution of salt, ground water tables, and porosity variations are particularly amenable to electrical prospecting methods. Geologic structures also can be indirectly detected by their effect on ground water.

Electrical resistivity surveys are conducted by generating a current in the ground between two electrodes and then using two other electrodes to measure the distribution of potentials (voltages). Different electrode configurations and spacings are used to detect lateral and vertical changes in resistivity for selected target depths. The measured variations are analyzed to establish probable cause. Induced polarization techniques measure the buildup or decay of induced voltages when the current is cycled on-off or alternated. Electromagnetic methods measure the electromagnetic field resulting from currents induced by an externally applied electromagnetic field. Electromagnetic surveys can be conducted from low-flying aircraft as well as on the ground and are therefore sometimes used in reconnaissance surveys. The Earth also exhibits spontaneous or natural voltages that can be measured and interpreted to investigate subsurface conditions and structure. These voltage variations may result from differences in fluid properties, the flow of ground water, mineralization, and chemical interactions. Telluric currents are sometimes used to investigate large-scale crustal features and lithologic variations.

Seismic exploration methods are perhaps the most useful geophysical tools for obtaining accurate representations of the subsurface geology at individual sites (227, 232, 233). They rely on the reflection or refraction of seismic (acoustic) signals due to contrasts in velocity or acoustic impedance (the product of seismic velocity and rock density). Acoustic signals are usually introduced into the earth by explosive sources, and vibrating or impacting masses. Seismic reflection surveys are particularly useful in mapping the attitude and continuity (or lack thereof) of subsurface rock beds. Detailed analyses of seismic reflection data have detected lateral variations in acoustic properties within individual beds and indirectly inferred the presence or the absence of hydrocarbons. Such field parameters as type, intensity, and frequency of the energy source and geophone spacing and filtering as well as data reduction methods are selected for individual surveys based on the depths, dimensions, and local conditions of the features of interest. In the petroleum industry, these parameters are often defined to provide information from depths of more than 1,000 meters (233). Special field parameters and techniques (high-resolution seismic) are available to explore accurately the shallower depths of interest for repositories. Seismic refraction is used less extensively than reflection, but it has application in evaluating near-surface features and crustal structure, as well as in establishing velocity profiles as a function of depth to aid in the design of local reflection surveys.

Geophysical logs in well bores are a powerful tool for correlating and interpreting subsurface geologic conditions, including the condition and fluid content of subsurface rocks (227). They supplement cores and rock samples and furnish a vertically continuous record of certain physical properties for each borehole. Many types of logs are used. Focused resistivity logs provide a reliable measure of in situ rock and fluid characteristics. Microresistivity logs measure the properties of small volumes of rock just behind the borehole wall and thus permit the boundaries of permeable and/or electrically resistive formations to be sharply defined. Gamma-ray logs indicate the clay content of various formations and are valuable in making lithologic interpretations in clastic rock sequences. Neutron logs are useful for identifying porous rock strata and rock densities. These logs respond mainly to the hydrogen content of the formation and indicate the presence of water, oil, or hydrogen-bearing minerals. Acoustic logs measure the velocity of sound and can also help determine the porosity of a formation. Other logging methods are available to supplement those listed above.

Improved borehole geophysics surveying methods are currently being developed whereby electrical, ultrasonic, or radar energy is generated by a downhole instrument, and the response is observed either at the surface or at depth in a nearby hole (234, 235). Preliminary results for special cases demonstrate a capability to resolve features measured in centimeters at distances of hundreds of meters and features measured in meters at distances of kilometers (230). Small features that perturb the movement of energy through the rock mass are identified by comparing the observed response to that expected if no perturbations were present. These methods have potential for improving the capability to resolve small features in the vicinity of a drill hole.

In summary, a wide variety of geophysical techniques is available for site selection and characterization. The selection and application of appropriate techniques depend on the geologic setting and the particular geologic problem being addressed. Interpretations of the resulting data are often difficult and are best made in conjunction with other geophysical surveys. Geophysical data must always be evaluated in light of existing geologic knowledge. However, geophysical surveys are a well established part of exploration prospecting, and proper evaluation can provide extensive information about subsurface geologic conditions. Such surveys are especially valuable in repository investigations because they permit investigation of subsurface conditions without extensive drilling.

II.D.4.4 Hydrologic Studies

The role of hydrologic studies in site exploration can be separated into three overlapping areas: (i) two-dimensional characterization of the surface and ground water systems for the region or hydrologic basin in which the site is located, (ii) three-dimensional characterization of ground water conditions at candidate sites, and (iii) computer analysis of the potential effects of the repository, the climate, or other perturbations on the ground water system.

It is necessary to determine whether ground water in the region of a site is static or flowing. If it is flowing, the direction and rate need to be determined. Ground water flows along a hydraulic gradient from regions of high to low hydraulic head (236). The flow rate generally can be determined provided the hydraulic gradient, effective porosity, and permeability can be measured at appropriate locations in the area of interest.

For unconfined ground water, the hydraulic gradient can be determined by mapping the distribution of hydraulic head (the potentiometric surface) as indicated by static water levels in wells and the elevation of perennial surface water bodies. In confined aquifers, the potentiometric surface must be determined by isolating aquifer intervals in wells and measuring either the hydrostatic pressure or the level to which water rises naturally (237, 238). Often, but not always, the unconfined static water table gives a reasonable approximation of the potentiometric surface of underlying aquifers.

Permeability, effective porosity, and rock compressibility can be determined by pump or injection tests in wells at the depth intervals of interest. Hydraulic properties are routinely measured for laboratory specimens of core or other rock samples obtained from the site (239). Using appropriately spaced wells, hydraulic communication between them can be established during pump or injection tests (236) to provide reliable calculations of in situ ground water velocities. If such tracers as tritium or fluorescent dyes are injected into upgradient wells, the timed appearance of the tracers in downgradient wells can be used to determine effective porosity and thus in situ velocities. Artificial gradients are usually established by pumping to draw water levels down or injecting water under pressure because natural ground water velocities are so low in many instances that the migration of a tracer to an observation well would require a prohibitively long time (as long as a year or more).

The isotopic dating of ground water (240-244) provides an alternative reference for evaluating calculated velocities. Water can be sampled for dating from selected discharge points and well locations throughout the ground water basin considered likely to be influenced by a repository. Differences in water ages among sampling points are used to calculate natural velocities. Careful analysis of these dates, with due consideration of the "age contamination" that can result from the mixing of waters of different ages (244), allows a general assessment of ground water residence times throughout a basin.

The identification and analysis of hydrologic conditions in nearly impermeable rocks is necessary to establish the degree of impermeability possessed by the host rock unit (245). Pulse injection tests aid in determining permeability in low-permeability rocks (246). Moreover, pressure decay curves for gases pressurized at selected borehole intervals can be used to calculate the permeability of the very tight rocks expected at repository horizons. Although present measurement techniques for hydraulic conductivity in nearly impermeable rocks may be in error by up to a few orders of magnitude (244), even the higher, most conservative values indicate that water moves extremely slowly in these rocks. Site-specific in situ testing may be required to narrow the bounds for estimates of in situ host rock permeabilities.

For rocks that possess a natural fracture system (e.g., granite, basalts, some shales, limestones, sandstones) the determination of near-field flow mechanisms is also evolving. Because fracture networks are not random, their nature and orientation within the system will be statistically determined. Methods designed to assess fracture effects on hydrologic flow are currently being developed at the Nevada Test Site (247), the Stripa mine in Sweden (212), and the Los Medanos site in New Mexico (248). The direct determination of hydrologic parameters in fracture networks includes conventional pump testing with multiple-point piezometers, tracer studies, and radioactive flow meter tests performed in wells or subsurface facilities constructed at the repository site or in rock bodies that provide a close analog of site conditions. Information about potential natural ground water communication between aquifers via faults, fracture systems, breccia pipes, and other structural elements that may influence the hydrology of the site is also evaluated by field studies and hydraulic tests.

Water influx at mines in crystalline rocks is a well-known phenomenon. However, where permeabilities are very low, mine ventilation commonly evaporates and removes most, if not all, of this water (212). Thus, the mines are usually "dry," although a small amount of water may continually flow into them. By sealing a room with airtight bulkheads and circulating controlled quantities of warm air, the amount of seepage water can be determined by measuring the humidity and mass of the circulating air. Data on fluid gradients around the sealed off chamber permit calculations of nearby rock permeabilities. Such an experiment, believed to be the first ever conducted, is being performed at the Stripa mine in Sweden (212, 249). (See also II.F.2.2.)

Data from hydrologic testing are combined with geologic interpretations of a site and its region to analyze the hydrologic flow system (II.F.1) (Figure II-11). A detailed three-dimensional model of the near field, including fracture-flow conditions as required, is then integrated with thermal-mechanical models to calculate the near field disposition of the wastes should they escape containment. The near-field models determine the source terms for regional two-dimensional flow models of a subject hydrologic basin. These regional models are used to calculate the isolation potential of the far-field natural system. Retardation mechanisms (e.g., sorption, precipitation and diffusion into the rock matrix) and radioactive decay chains for the radionuclides will be factored into both near and far-field models of the isolation system. Conservative assumptions regarding potential changes in the hydrologic system that may be caused by climatic and tectonic changes will be used to develop scenarios for modifying models of present ground water flow conditions and calculating the effects of the assumed changes.

The potential for climatic changes and the attendant effects on the hydrologic regime at each site are evaluated. Palynology (study of fossil spores, pollen, and seeds) provides data concerning the past distribution of floral assemblages (171, 250, 251). By establishing the present relationship between climate and vegetation patterns, one can estimate past climates from the character and distribution of past vegetation. In unconfined ground water systems, the effects of ancient water tables on minerals may be preserved, providing a means for determining past water levels. The effects of climatic changes on the hydrologic system are complex. The hydraulic gradient is dependent on the mass balance between recharge and discharge (236). If recharge increases--for example, during a pluvial episode--hydraulic gradients and water levels may or may not change, depending on the ease with which the flow system can transmit the perturbation to discharge regions.

In summary, hydrologic studies represent the point at which all data on the natural system are integrated to assess by numerical models the ability of the natural system to provide the required isolation. Numerous hydrologic field data and test results are required. The techniques for obtaining most of them are currently available; others, including improved techniques for ground water dating, fracture-flow modeling, and permeability determinations for low permeability rocks, need development (203, 242, 244, 245). Hydrologic models combined with geochemical studies are used to estimate the likely composition and concentration of any and all radionuclides at any given point and time relative to a site's regional aquifer system (see II.F.2). When applied, they assist in determining whether a mined geologic repository at a particular location can provide the required isolation (II.A.1, Objective 2). Although the development of a detailed, accurate hydrologic model for each site will require considerable time, bounding assumptions about hydrologic parameters can be applied during the screening process to assess the general quality of hydrologic systems and to identify areas requiring better definition.

II.D.4.5 Laboratory and Field Testing

The thermal properties of potential host rocks are measured in the laboratory by accepted methods (150, 151). Standard-sized cylindrical specimens are subjected to a controlled thermal power source at one end; increasing temperatures and dimensions are measured either along the axes or along the outside lengths of the specimens. The results are used to calculate volumetric expansion coefficients and thermal conductivity. The specific heat of a rock is determined by standard calorimetry (150).

Mechanical properties of potential host rocks are also measured in the laboratory by standard techniques and apparatus (151); the results are used in preliminary models of the repository's response and to help determine which properties require better definition by field testing (158). The compressive strengths of potential host rocks are determined in accordance with well-accepted methods by observing which states of stress and temperature cause fracturing. Both uniaxial and triaxial compressive tests are used. After the critical strength is determined, tests are run that gradually increase stress to just below the rupture point and then gradually release the stress to original conditions in order to observe whether permanent deformation will occur in slowly pulsed stress environments below the critical strength (252). Standard tests are also performed to determine the tensile strength of rocks, although it is generally conceded that large, fractured rock bodies have a very low tensile strength.

Rocks that exhibit significant creep behavior within the temperature ranges expected in radioactive waste repositories are tested in triaxial testing machines that measure creep deformation at various combinations of stress state and confining pressures (152, 162). Measurements are made over periods of up to several months, and both primary and secondary, if exhibited, creep behavior is observed.

Synergistic effects between thermal and mechanical properties are determined for laboratory samples by obtaining, for example, hysteresis data on mechanical response as a function of rock temperature or obtaining thermal conductivity data as a function of lithostatic stress. The effects of the rock's fluid content on specific heat, critical stress, and thermal conductivity are also being investigated.

In measuring sorption capacity, water doped with radionuclides is passed through a rock and the amounts of radionuclides in the rock and the water are measured by radiography (scintillation counts) and, if possible, by chemical separation. Transient and batch methods for determining sorption are currently being standardized (147). Techniques are also being developed for pinpointing mineralogic and molecular affinities for sorbed radionuclides, allowing a better understanding of the materials and mechanisms responsible for the sorption process.

Because questions arise (211, 244, 253-255) regarding the validity of using thermal, mechanical, and sorption data obtained at laboratory scales (generally measured in centimeters) to predict behavior in a complex, heterogeneous, and probably anisotropic environment, the Department has a research program for conducting field determinations of thermal, mechanical, and chemical behavior under expected repository conditions. Tests performed to date are more fully addressed in Section II.F.1; only the general methods for obtaining data from these tests are discussed here. Data from field tests can be a very useful in supporting professional judgments about model results in particular and the suitability of repository sites in general.

Field tests, whether near the surface, in existing mines, or in newly constructed mined cavities, generally involve single or multiple heat-sources, emplaced in drill holes, and measuring instruments in an array surrounding the heat source. Heat can be generated by electric heaters or spent-fuel assemblies. Heaters used to date have operated at power inputs ranging from about 1 kW to 5 kW, simulating spent fuel less than 10 years old (see II.F.1).

Instrumentation arrays are designed to monitor heated temperature, rock deformation, and, in some tests, water content, chemistry, and rock stress as functions of time and distance from the heat source. If spent fuel is used as the heat source, radiation may also be monitored. Temperatures are measured as a function of time by thermocouples deployed on the surface of the heater, at the rock wall of the emplacement hole, and in separate monitoring holes at various distances from the heat source. Rock deformation from thermal expansion is monitored by extensometers (256-258). These devices measure slight changes in length, across a distance of meters, by a variety of techniques. As with thermocouple data, extensometer measurements are telemetered to a computer that provides a three-dimensional representation of rock deformation as a function of time. In addition to the more common linear transducer and vibrating wire strain meters, laser interferometry (247) is a developing means for detecting the minute displacements expected in field heater tests. Precisely surveyed laser sources and reflection mirrors allow interference patterns from the source and reflected light to be observed and displacements calculated.

Water content, gas generation, and chemical properties around a heat source are monitored with humidity gages, pressure gages, pH meters, and moisture and gas traps (212, 247, 248). Seepage, pressure, humidity, and chemistry along cracks in the rock or in nearby monitoring holes are measured to determine the effects of thermal loading on the fluid system in the rocks.

Natural rock stress is measured in field tests by overcoring and flat-jack techniques (151, 211). Overcoring allows stress to be calculated from comparing shapes of the core before and after stress is released by removing of the core from the subsurface. Flat-jack techniques utilize a pressure-sensitive plate inserted into the rock mass to determine the natural, pretest stresses and to monitor any changes during the propagation of the thermal pulse during the test. Natural stress can also be measured by hydraulic fracturing (259), whereby fluids are injected into wells under controlled pressure. The pressure at which sudden pressure drops occur is used to calculate the minimum principal in situ stress for the depth interval being tested.

Determination in the field of the synergistic effects of thermal, mechanical, and fluid behavior is a multivariant problem. Chapter II.F addresses the rationale of using the results of small-scale laboratory experiments to design field and large-scale laboratory tests whose results are used, in turn, to design in situ tests at selected repository sites. The current and evolving measurement capabilities can provide data for constructing and, in an iterative process, improving models for predicting the response of the natural system to heat-generating radioactive wastes.

II.D.4.6 Geologic Forecasting Studies

The field of geologic forecasting is less well developed than the other topics discussed so far (244, 253, 255). Geologic research has largely concentrated on characterizing present-day natural processes and events and on historically reconstructing the distribution, magnitude, and sequence of past events. However, it is possible to make some general observations about the methods for investigating future tectonic activity, including volcanic eruptions, folding, epeirogeny, fault movements, salt diapirism, and seismic activity.

Both the likelihood and the consequences of changes in the natural system with regard to containment and isolation need to be estimated. For natural events, estimates of the likelihood for recurrence are based on the geologic principle of uniformitarianism, which generally states that the basic processes responsible for past events are determined by physical and thermomechanical laws and thus will continue into the future (206, 260). For this reason, a study of the history of geologic events is utilized to establish a hypothesis regarding the distribution of causative processes in time and space and, in turn, projecting the continuation of that distribution into the future.

Plotting space-time relationships of past events allows a calculation of past rates and distributions of occurrence for tectonic events (182, 183, 261, 262). The probabalistic extrapolation of these rates into the future must be weighted against deterministic tectonic models such as plate tectonics to determine whether observed space-time distributions are likely to continue or be modified. The geographic scale for which data are compiled is of critical importance and needs to be evaluated. In general, for larger areas consensus is more readily obtained among earth scientists about tectonic processes. Conversely, averaging probabalistic projections for individual events over large areas decreases their reliability for a given site. Thus, a reasoned interpretation of probabilistic and deterministic approaches is required to assess the likelihood of tectonic events that might disrupt a repository's natural system. This combination of methods is most developed for assessing seismic hazards (177, 261-265).

Potentially active faults can be deterministically identified from geologic, geophysical, seismic, and natural stress data. Standard earthquake-hazard assessment provides probabilistic estimates of expected return periods at specific sites for ground motions of various magnitudes. These methods are used in conjunction to help determine appropriate seismic design requirements. Similar methods are evolving for volcanic and diapiric phenomena.

The consequences of tectonic events must also be estimated. Observations of earthquake-related damage, both at surface facilities (266) and in mine tunnels (173, 176, 267, 268), provide empirical data for substantiating calculations based on the physical response properties of the structures of interest.

The consequences of such intrusive processes as salt diapirism and volcanism are estimated by studying the geometry, disruption zones, and chemical alterations associated with existing intrusions. The movement of faults, intrusions of material, and tectonic epeirogeny are evaluated also in terms of their effects on hydrologic systems and erosional processes where conditions allow current study. The principles gleaned from analog observations obtained away from repository areas are applied to conditions existing at potential repository sites. Impacts of faulting, erosion, and intrusion are estimated parametrically by assuming various event scenarios and analyzing their effects on the hydrologic flow models.

In summary, the prediction of tectonic events and their potential impacts over periods of tens of thousands of years is an advancing capability. The unique requirements of radioactive waste management have generated the first demands for applying long-term geologic predictions. However, careful selection of repository sites can reduce the likelihood of tectonically induced disruptive events to almost zero, based on an uniformitarian assumption. The potential impacts of postulated events will be defined by scenario analysis in order to assess their effects on containment and isolation.

II.D.4.7 Resource Studies

The potential for exploiting mineral, energy, water, and subsurface land-use resources both now and in the future will be assessed throughout the site-selection process. Geologic, geophysical, borehole, and geochemical studies conducted during site exploration and qualification provide data for evaluating the potential for resource development. The exploration and ultimate selection of a repository are the converse of seeking an ore body or an oil field, in that investigations are conducted to locate areas with a low resource potential. However, the proper resource characterization of the region is accomplished through the approaches and studies already outlined. With regard to the long-term potential for resource development, geochemical data for all elements and compounds can be compared with the average crustal concentrations. If any characteristic, including thermal gradients, in the site location significantly exceeds the crustal average, its potential value to future generations needs to be carefully considered. The consequences of inadvertent human intrusion into the repository due to resource exploration at some future time must also be considered (see II.E.3 for a discussion of human-intrusion issues).

II.D.5 Status of Ongoing Exploration Programs

The preceding sections of Chapter II.D have described the factors of the natural system important in site selection, design, and construction of deep geologic repositories; the requirements that must be satisfied by a repository site; and the methods available or being developed for characterizing and assessing the natural system.

This section summarizes the status of geologic investigations conducted by the Department over the last several years. The site selection process, described in Section III.C.1, is generally conducted in four steps: national screening surveys, whose objective is to identify places that have some potential for waste isolation; regional studies, which evaluate a specific region of interest; area studies, which are conducted to characterize the areas of interest described by the regional study; and location studies, which further narrow the scope of the investigation to a site or sites.

Individual investigations are in various stages of the site selection process. Descriptions are given of the status of current investigations in (i) the Gulf Interior Region Salt Domes, (ii) the Paradox basin, (iii) the Permian basin, (iv) the Salina basin, (v) basalt flows at the Department's Hanford Site, (vi) the Department's Nevada Test Site, and (vii) planned future investigations.

Additional information regarding the technical scope and current status of the studies described in this section is presented in Appendix B to this statement. Appendix B repeats portions of the summary provided in this section. Many geologic time periods are referred to by name in this section; therefore a geologic time scale is presented (Figure II-12) to enable the reader to associate named geologic periods with historical age (269).

II.D.5.1 Gulf Interior Region Salt Domes

The Gulf Coastal Plain of Texas, Louisiana, and Mississippi, and adjacent offshore areas contain more than 500 salt domes, 263 of which are known or suspected to be on land. These interior domes were evaluated in 1963 by the U.S. Geological Survey, and 36 were identified as potentially acceptable for repository siting (270). The USGS screening criteria were (i) depth to salt of less than 2,000 ft and (ii) lack of previous use (oil, gas, sulfur, or brine mining). After the USGS study, the Department, with the participation of USGS district offices, selected 125 interior domes for detailed studies which provided important background data for current investigations (271-275).

ERA	PERIOD	EPOCH	YEARS DURATION	YEARS BEFORE THE PRESENT
CENOZOIC	Quaternary	Holocene	To Present	
		Pleistocene	1,000,000	1,000,000
	Tertiary	Pliocene	12,000,000	
		Miocene	12,000,000	
		Oligocene	11,000,000	
		Eocene	22,000,000	
		Paleocene	5,000,000	63,000,000
MESOZOIC	Cretaceous		72,000,000	135,000,000
	Jurassic		46,000,000	181,000,000
	Triassic		49,000,000	230,000,000
PALEOZOIC	Permian		50,000,000	280,000,000
	Pennsylvanian		30,000,000	310,000,000
	Mississippian		35,000,000	345,000,000
	Devonian		60,000,000	405,000,000
	Silurian		20,000,000	425,000,000
	Ordovician		75,000,000	500,000,000
	Cambrian		100,000,000	600,000,000
PRECAMBRIAN				

Figure II-12. Geologic Time Scale

Source: (Reference 269) A. Holmes and L. Holmes, Principles of Physical Geology, Third Edition, Halsted Press, New York, NY, 1978

Since 1978, interior domes have been evaluated by the Department in terms of geologic and other screening specifications (188, 276, 277). The three criteria that dominated the screening were (i) the top of the domes should be at depths of less than 915 m, (ii) domes should have cross-sectional areas of more than 1,000 acres, and (iii) domes should not have been used for hydrocarbon production or storage, or any other mineral-related activities. This latest study resulted in the selection of eight salt domes for the "area" study phase (see Section III.C.1). These domes are distributed among the States of Louisiana, Mississippi, and Texas as follows: Vacherie and Rayburns (Louisiana); Richton, Cypress Creek, and Lampton (Mississippi); and Oakwood, Keechi, and Palestine (Texas). The Palestine dome was dropped from further consideration in 1979 because of hydrologic uncertainties related to earlier solution mining (278). Fifteen karst-like collapse structures over the Palestine dome have been attributed to extensive brine production and solution collapse. Although the solution mining occurred from 1904 to 1937, three collapses have occurred since, one as recently as 1978.

Investigations under way at the remaining seven domes include hydrologic studies of the three sedimentary basins in which the domes occur (Figure II-13) as well as dome-specific geologic and hydrologic studies. Aquifers are being investigated to depths of 2 km in each basin by borehole pump tests. Geologic studies include regional and dome-specific field mapping with emphasis on evaluating Quaternary terraces, remote-sensing data, geophysical well logs, and deep seismic data. Understanding of dome locations is being refined by gravity surveys, high-resolution seismic reflection and refraction surveys, and borehole evaluations.

Drilling of a total of 34 deep exploratory holes in three states aggregating 89,189 ft and drilling of additional intermediate-depth and shallow holes produced no evidence to disqualify any of the remaining seven domes, although several characteristics need careful evaluation against the siting criteria. Exploration is proceeding at a variable pace for all seven domes; only the Lampton dome in Mississippi and the Keechi dome in Texas have not been explored by drilling or by seismic methods.

Studies of hydrologic stability are centered on determining the resistance of salt masses to external dissolution. Current evidence suggests

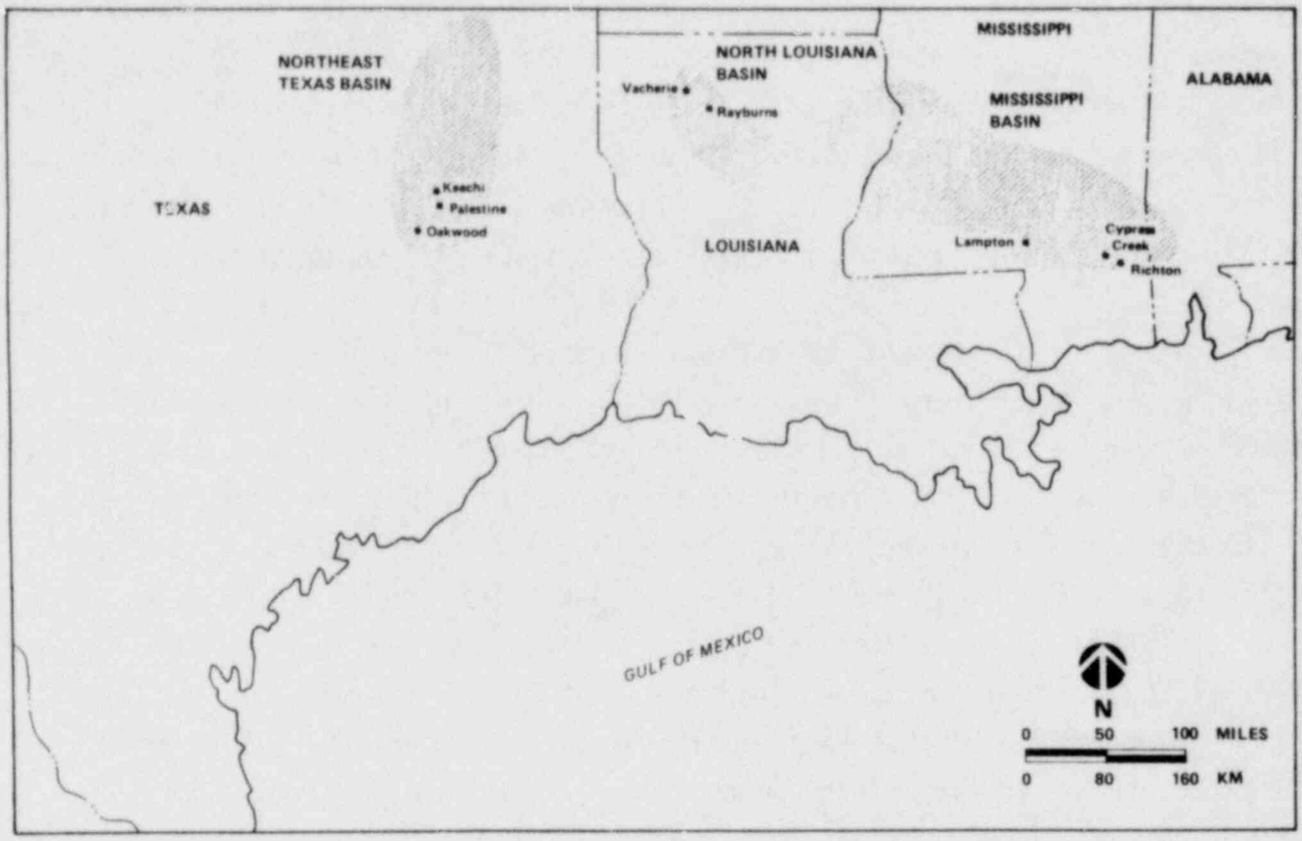


Figure II-13. Gulf Interior Region Gulf Coast Salt Domes Recommended for Further Study by the U.S. Geological Survey

Source: (References 270 and 275) Adapted from

R.E. Anderson, D.H. Eargle, and R.O. Davis, Geologic and Hydrologic Summary of Salt Domes in Gulf Coast Region of Texas, Louisiana, Mississippi, and Alabama, Open-file Report USGS-4339-2, U.S. Geological Survey, 1973; and

J.O. Ledbetter, W.R. Kaiser, and E.A. Ripperger, Radioactive Waste Management by Burial in Salt Domes, p. 82, AEC Contract AT 40-1-4639, Engr. Mechanics Research Lab., University of Texas, Austin, TX that salt domes have become, through geologic time, encapsulated in plastic clays or other impermeable minerals. These layers of impermeable sediments appear to have prevented the domes from dissolving since their formation 25 to 30 million years ago. Direct evidence for this hypothesis is found in oil company core samples and in geologic and drilling logs. The degree of continuity of the encapsulating clay needs to be determined. Preliminary calculations suggest long tra times for radionuclide migration to the accessible biosphere (at least 100,000 years). This is because thick sections of clay with a very low hydraulic conductivity (permeability) occur around and over the domes.

With regard to tectonic stability of the domes, studies of Tertiary and older strata suggest that movement of the salt in the Gulf interior region ceased perhaps as long as 30 million years ago (272, 279). Assessments based on the thinning of sediments over the dome show a slow and declining rate of growth through geologic time, with values in the early Cenozoic well below 0.2 mm/yr (272). In addition, careful investigations of the stratification of Quaternary units in Louisiana have revealed no characteristics attributed to dome growth. All of the seven domes being investigated are considered to be tectonically stable. No capable faults* are known to exist in the vicinity of the domes.

There is minor, and declining, oil production at the Oakwood and Cypress Creek domes. All domes have one or more resource exploration holes that penetrate to the salt. All domes have some degree of caprock.

In 1980, two or three domes will be recommended for further examination in the "location" study phase (see Section III.C.1) of the site exploration process.

---

*Capable faults - As defined by the Nuclear Regulatory Commission in 10 CFR Part 100, Appendix B - "capable fault" is a fault which has exhibited one or more of the following characteristics: (1) Movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years; (2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault; (3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

II.D.5.2 Paradox Basin

The Paradox basin in southeastern Utah and southwestern Colorado is a part of the Colorado Plateaus, an area of high plateaus and deeply incised canyons. The extensive flat-lying sedimentary rocks are sometimes folded into structural upwarps and occasionally interrupted by Tertiary volcanoes, volcanic necks, and igneous intrusions. Approximately 12,000 square miles of the Paradox basin are underlain by the bedded salt of the Paradox Formation, which was deposited about 300 million years ago during Pennsylvanian time. More than 25 salt layers, separated by interbeds of shale, carbonates, and anhydrite, are present in some parts of the basin. The Paradox basin is one of the salt regions identified during a national screening as having potential for the eventual siting of a waste repository (280). Evaluation of the basin was begun in 1972 by the U.S. Geological Survey (281).

Existing information on the Paradox basin is not yet sufficient for assessing the suitability of individual parts of the region for a repository. Current investigations are acquiring data on the hydrologic and chemical characteristics of the ground water flow systems, Quaternary history, and present-day seismicity for the basin in general and for four particular areas recommended for further study. These four areas were selected on the basis of geologic factors (e.g., depth from Earth's surface to salt, thickness of salt, and location of mapped faults) and environmental screenings (see Figure II-14). One of the four areas, Salt Valley, in Grand County, Utah, had been previously identified by the U.S. Geological Survey (281). The other three areas, Gibson Dome, Elk Ridge, and Lisbon Valley, are in San Juan County, Utah. The results of studies now in progress will be used in deciding whether locations in one or more of the areas will be recommended for further investigation.

In the near future at least two deep holes, one in the Gibson Dome area and one in the Elk Ridge area, will be cored, logged, and extensively tested (282). Samples of fluids recovered during testing operations and samples of cores will be analyzed at the site and in the laboratory for water content of salt, mineral composition, kerogen and hydrocarbon content, and hydrology. A seismic reflection line is planned at Elk Ridge. Surface

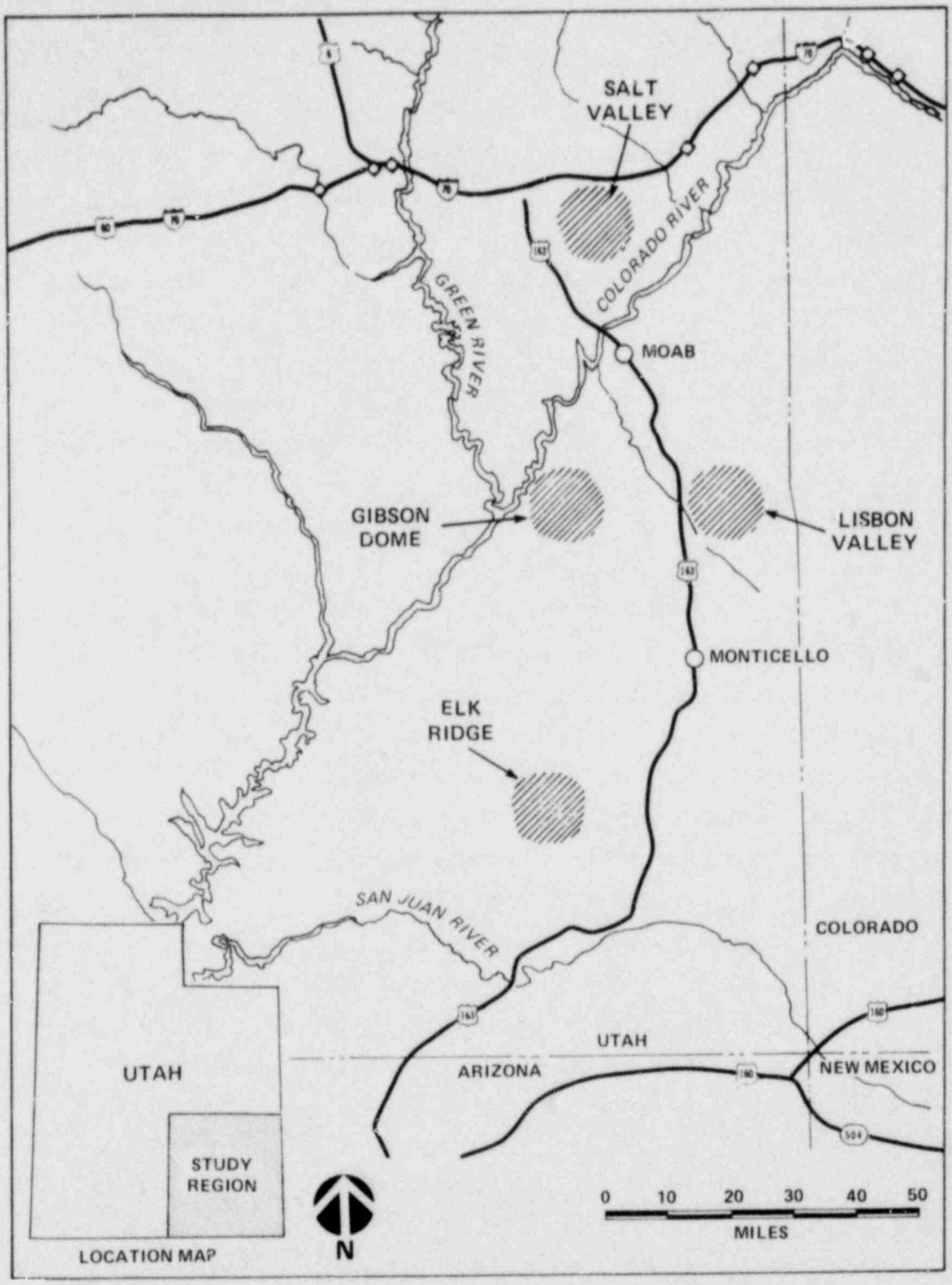


Figure II-14. Areas of the Paradox Basin Identified for Further Evaluation

Source: (Reference 281) R.J. Hite and S.W. Lohman, Geologic Appraisal of Paradox Basin Salt Deposits for Waste Emplacement, Open File Report 4339-6, U.S. Geological Survey, 1973 electromagnetic surveys will be conducted at Gibson Dome and Elk Ridge. Hole-to-surface electrical resistivity surveys will be run in the vicinity of the planned deep drill holes. Data collected over a period of 6 months by a 24-station microseismic network are being analyzed.

Area-level field investigations for Lisbon Valley will be much less extensive than those in the other three areas, because of the large amount of amassed earlier information for mining activities in this area. The potentially unfavorable characteristics of this area include the near proximity of a producing oil field and extensive uranium deposits and exploration (283). Moreover, the area is more geologically complex than the Gibson Dome and Elk Ridge areas but probably less complex than the Salt Valley area (284).

Preliminary results from investigations conducted to date indicate that bedded salt layers of sufficient volume are present at suitable depths in the Paradox basin. Historically, many earthquakes with Richter magnitudes exceeding 1.0 have occurred within 200 mile from the basin, but only 17 were in the basin itself. The largest had a Richter magnitude of 4.3 and was located near the basin margin. Potential resource-conflict and ground water flow system evaluations are in progress.

II.D.5.3 Permian Basin

The Permian basin is a series of sedimentary basins in which rock salt and associated salts accumulated during Permian time more than 200 million years ago. It includes the western parts of Kansas, Oklahoma, and Texas and the eastern parts of Colorado and New Mexico. Since Permian time the basin has been relatively stable tectonically, although some parts have tilted and warped, undergone periods of erosion, and been subjected to major incursions of the sea.

The Permian basin is one of the salt regions identified in a national screening as having potential for the siting of repositories (280). A regional characterization has been completed, and area-level studies are under way in the Palo Duro and Dalhart basins of Texas, described below in II.D.5.3.1.

In the New Mexico portion of the Permian basin, search for sites suitable for the Waste Isolation Pilot Plant led to the Delaware basin and the subsequent selection of a proposed site in the Los Medanos area, described in II.D.5.3.2.

II.D.5.3.1 Palo Duro and Dalhart Basins

The Palo Duro and Dalhart basins (Figure II-15) were identified as areas with potential for siting a waste repository through screening of the Permian basin bedded-salt deposits (285). These basins are located largely in the Panhandle of Texas. The area is typified by an almost featureless plain that is dissected by headward-cutting streams and is underlain by nearly horizontal Mesozoic and Cenozoic sedimentary formations.

Current investigations are in a subregional evaluation phase in which some field work has been done. Data available for direct analysis include (i) 8,000 ft of salt-bearing core, (ii) petroleum source-rock quality and thermal maturity data for resource assessment studies, (iii) drill-stem test data for regional hydrogeologic studies, and (iv) quantitative data on the climatic history, erosion potential, and shallow subsurface salt dissolution for predicting the long-term geomorphic integrity of the Texas Panhandle.

The data assembled to date are preliminary. Detailed area and site investigations began in FY 1980. Specific questions pertaining to hydrology, tectonics, geology, and resource evaluations will be the subjects of proposed investigations (286). However, based on (i) the abundance of evaporite deposits which are interbedded with potential barriers to ground water migration (e.g. shale), (ii) the remoteness of the region, and (iii) its marginal potential for resources, the region is a likely candidate for further evaluation.

II.D.5.3.2 Los Medanos Site

Geologic and hydrologic investigations of a site have been under way in Southeastern New Mexico since 1972. This site is within a region of the Delaware bedded-salt basin and is about 26 miles east of Carlsbad, New

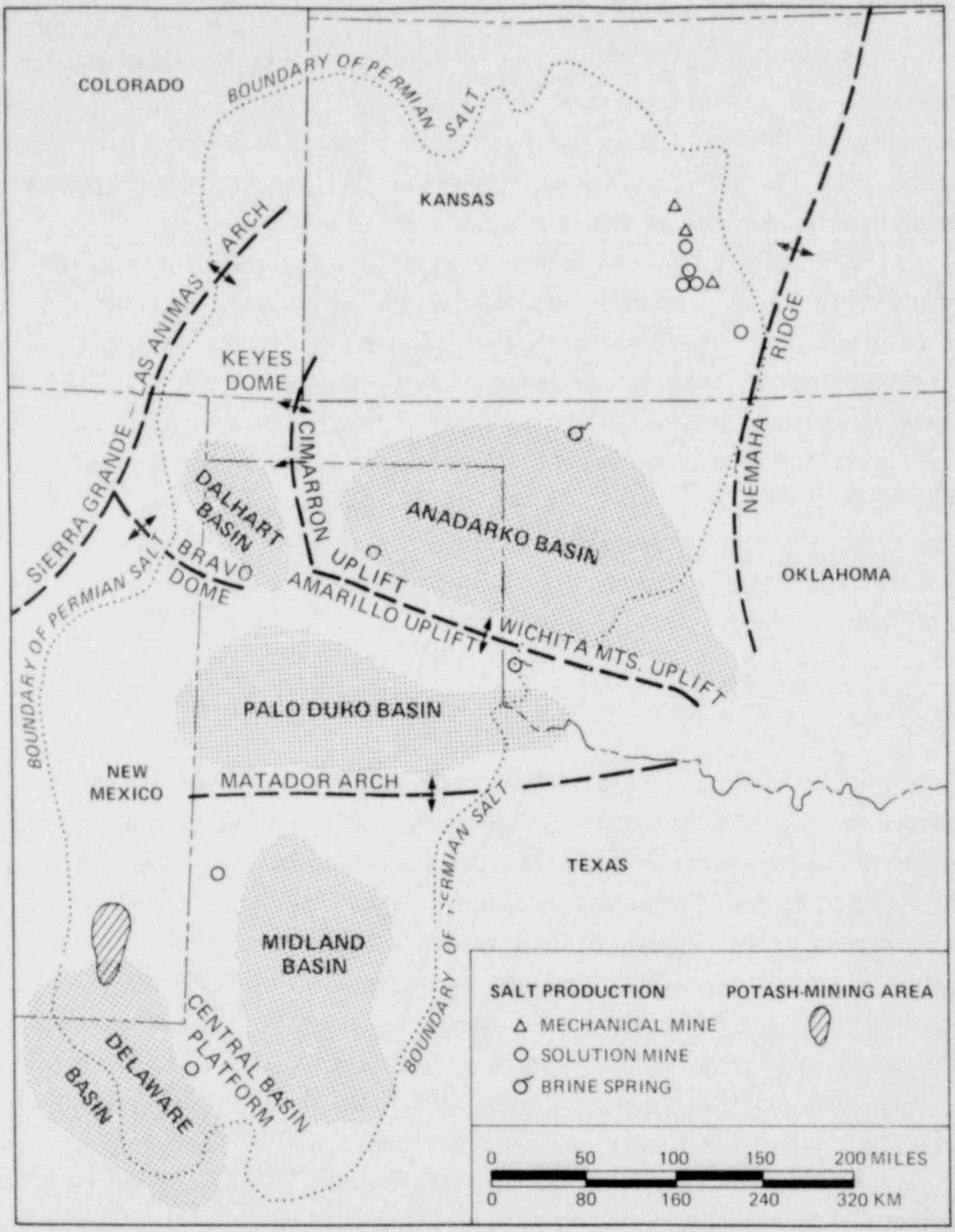


Figure II-15. Map of Permian Basin Salt Area Showing Principal Structural Features

Source: (Reference 280) K.S. Johnson and S. Gonzales, Salt Deposits in the U.S. and Regional Characteristics Important for Storage of Radioactive Waste, Y/OWI/Sub-741411, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

Mexico (Figure II-16). This site is well characterized and has previously been proposed by the Department for the Waste Isolation Pilot Plant (WIPP), a facility which was authorized by Congress for the disposal of transuranic wastes from the defense program. The site investigations are summarized in a comprehensive Geological Characterization Report (192).

The data collected to date indicate that the site has all the required features, with the possible exception of some conflict with natural resources. While it is possible that future exploration at depth or improved understanding of geologic processes could reveal aspects undesirable for a repository, these prospects are unlikely. Calculations performed for a potential breach of the WIPP repository by future human activity, believed to be more likely than natural failures of the geologic barriers, indicate that, even under extremely conservative assumptions, the potential hazard to the general population is very slight--less than that from naturally occurring radiation.

II.D.5.4 Salina Basin

The Salina basin as here defined is a portion of the northeastern United States underlain by bedded salt in a rock unit called the Salina Group or Formation of Late Silurian age. This salt formation is present in the northern Appalachian and Michigan basins in parts of Michigan, Ohio, Pennsylvania, and New York (Figure II-17).

A regional study of the Salina basin has been made from the existing geologic literature and geologic data available from public and private sources. For the Ohio and New York portions of the Salina basin, the regional studies have been completed (287, 288) and initial evaluations have identified study areas that appear to be geologically favorable for more detailed field investigations. The Michigan portion of the Salina basin has not been studied in sufficient detail to allow the identification of study areas. However, it is known that Michigan has salt beds of sufficient thickness and extent to meet present specifications for waste repositories and that these beds occur at suitable depths.

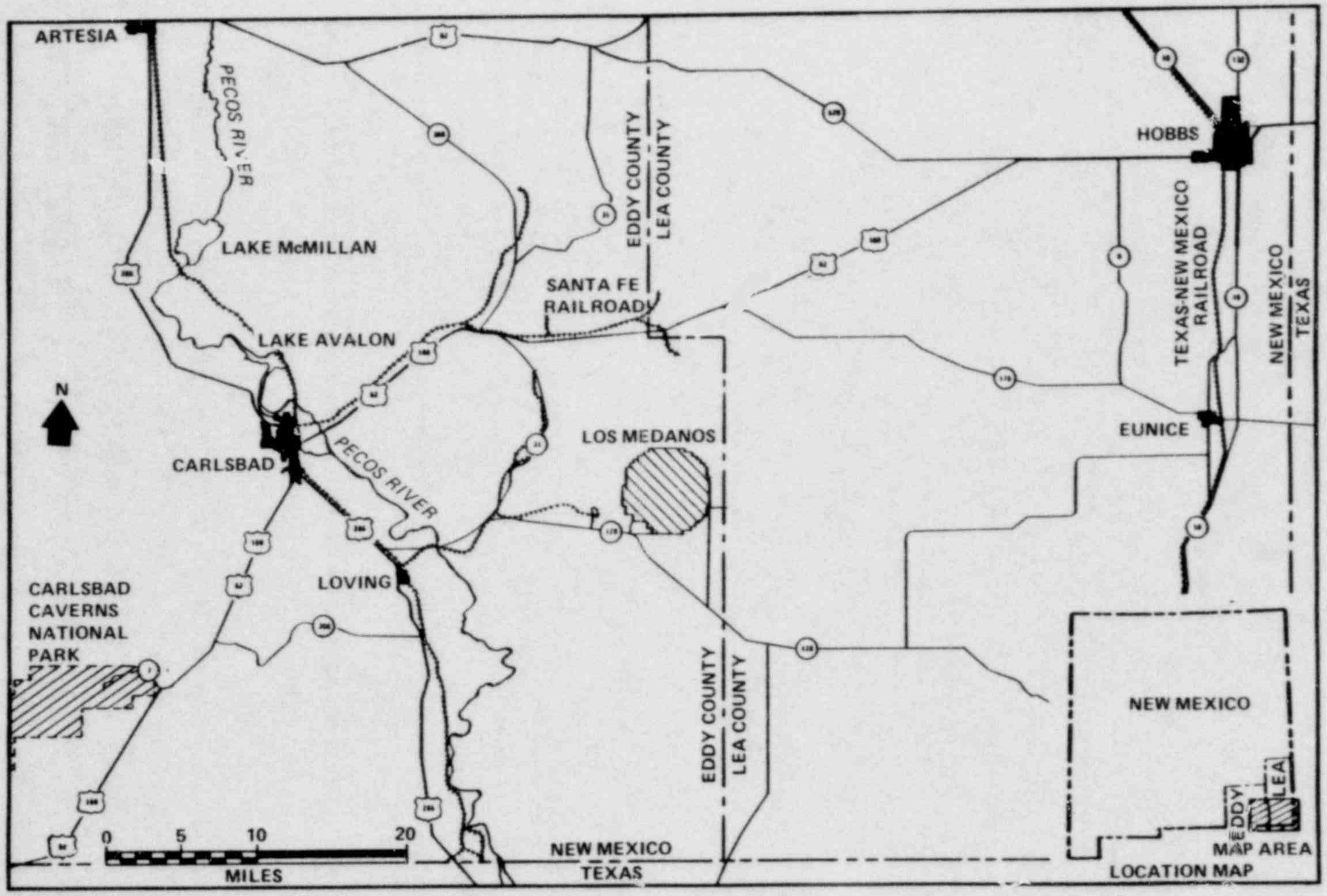
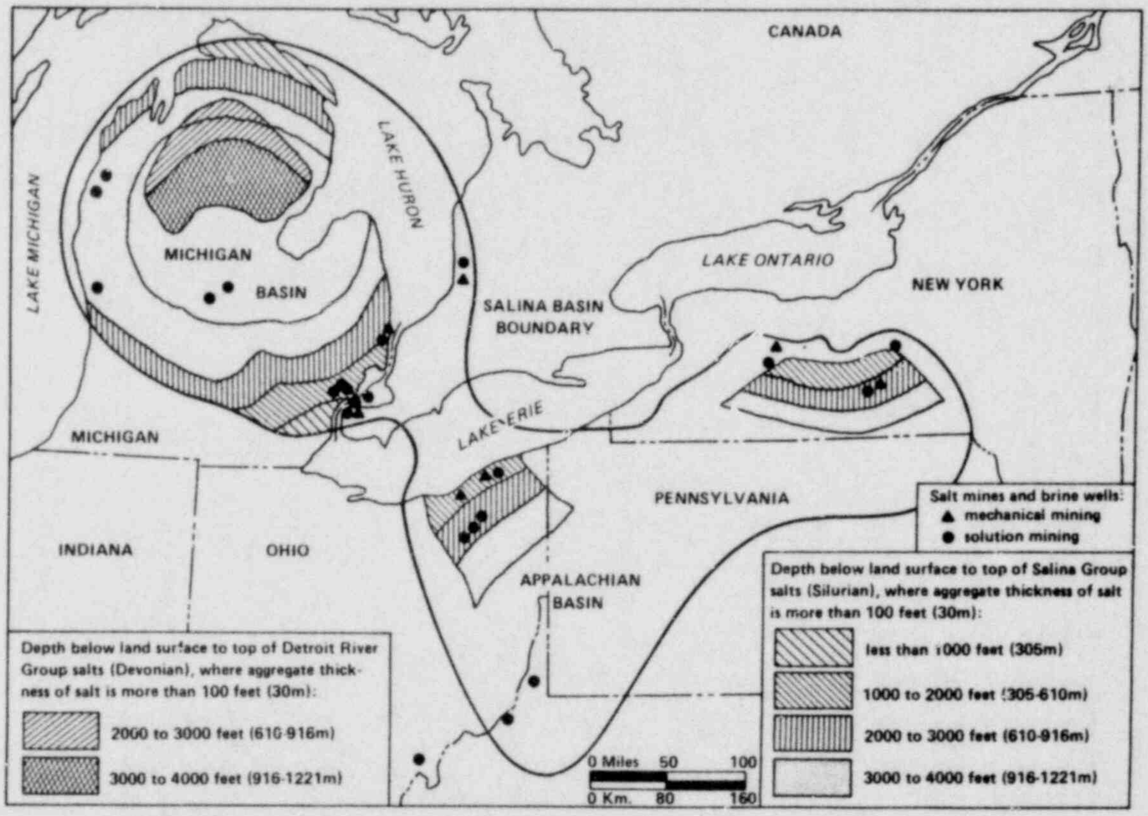


Figure II-16. Location Map for Los Medanos



MAP SHOWING DISTRIBUTION AND DEPTH BELOW LAND SURFACE OF SILURIAN AND DEVONIAN SALT DEPOSITS IN THE NORTHERN APPALACHIAN AND MICHIGAN BASINS OF MICHIGAN, OHIO, NEW YORK, AND PENNSYLVANIA.

Figure II-17. Map of Michigan and Northern Appalachian Basins (Salina basin)

Source: (Reference 280) K.S. Johnson and S. Gonzales, Salt Deposits in the U.S. and Regional Characteristics Important for Storage of Radioactive Waste, Y/OWI/Sub-741411, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

No field investigations have been carried out by the Department in the Salina basin. Some field work in support of repository siting has been done by the U.S. Geological Survey in New York and in Pennsylvania.

The Salina basin is considered to be tectonically stable; it has low seismicity and is far from crustal plate boundaries. The potential for uplift or subsidence in the next million years appears to be minor. Even at extreme rates of denudation, surface lowering by stream erosion will not threaten a repository located at least 1,000 ft below the surface. Although glacial scour in the Finger Lakes region of New York has reached estimated depths of 1,200 ft to possibly as much as 1,800 ft, this process is not general and was caused by specific geologic conditions. The amount of glacial scour in valley areas needs to be investigated further, but upland areas appear to be favorable with regard to potential future glacial erosion.

New York salt beds were deformed by Paleozoic thrust faulting, folding, and probable decollement with tear faults. The structure is complex, but present information indicates that large areas within the State should be investigated further.

A large area in Northeastern Ohio meets preliminary screening requirements: salt with an aggregate thickness of 75 ft or more, and depths between 1,000 and 3,000 ft. Ohio has more oil and gas production than New York and a higher potential for future resource development. Resource conflicts may be severe for siting a repository in Ohio.

Regional analyses suggest that Michigan is geologically favorable. No detailed screening has been done, but the stratigraphy and the structure of the salt beds appear to be promising.

Much additional information is needed before a repository site could be identified in the Salina basin. The deep ground water circulation and flow systems are not known. The detailed composition of the salt beds is not known. The water content and the mineral impurities of the salt need to be determined. In addition, the nature of facies changes within the salt beds and surrounding rocks needs to be evaluated. Dissolution of salt is probable in New York where the Salina Group crops out. The rates and processes of dissolution are not known and need to be evaluated. Northeastern Ohio does not appear to have significant present-day dissolution. The potential for oil and gas development in the Salina basin needs to be evaluated. The regional geologic studies of the basin indicate that New York, Ohio, and Michigan all have both favorable and unfavorable geologic characteristics for a repository. At the present, no part of the basin can be judged acceptable or unacceptable for repository siting.

II.D.5.5 Basalt Waste Isolation Project

The Basalt Waste Isolation Project (BWIP) is evaluating the Department's Hanford Site in the State of Washington to determine whether it contains a suitable location for a repository in basalt. The Hanford Site, 576 square miles in ar ., is in the center of the Pasco basin (Figure II-18). Geologic and hydrologic investigations have been under way since 1977 and represent a continuation of an effort conducted between 1968 and 1972. Most of the results of these investigations have been published (289, 290).

Current data and understanding indicate that basalt is a suitable medium for the disposal of radioactive waste. Investigations of the Hanford Site to date indicate that it has suitable geologic and structural characteristics. The location and the movement of water in the unconfined and upper basalt confined aquifers are well understood (291, 292). Questions about the location and movement of the water in the interbeds and interflows of Wanapum and Grande Ronde Basalts are being addressed (293) and should be resolved in the next 2 to 3 years. The tectonic conditions of the area have been investigated, and it appears that the area is sufficiently stable for siting a repository (294). Current information and understanding indicate no conflicts with natural resources.

II.D.5.6 Nevada Nuclear Waste Storage Investigations

The Nevada Nuclear Waste Storage Investigations (NNWSI) are evaluating the suitability of the Department's Nevada Test Site (NTS) in Nye County, Nevada (Figure II-19). Because waste isolation activities must not interfere with the prime mission--nuclear weapons testing--exploration for a suitable repository site on the NTS is currently limited to the southwest

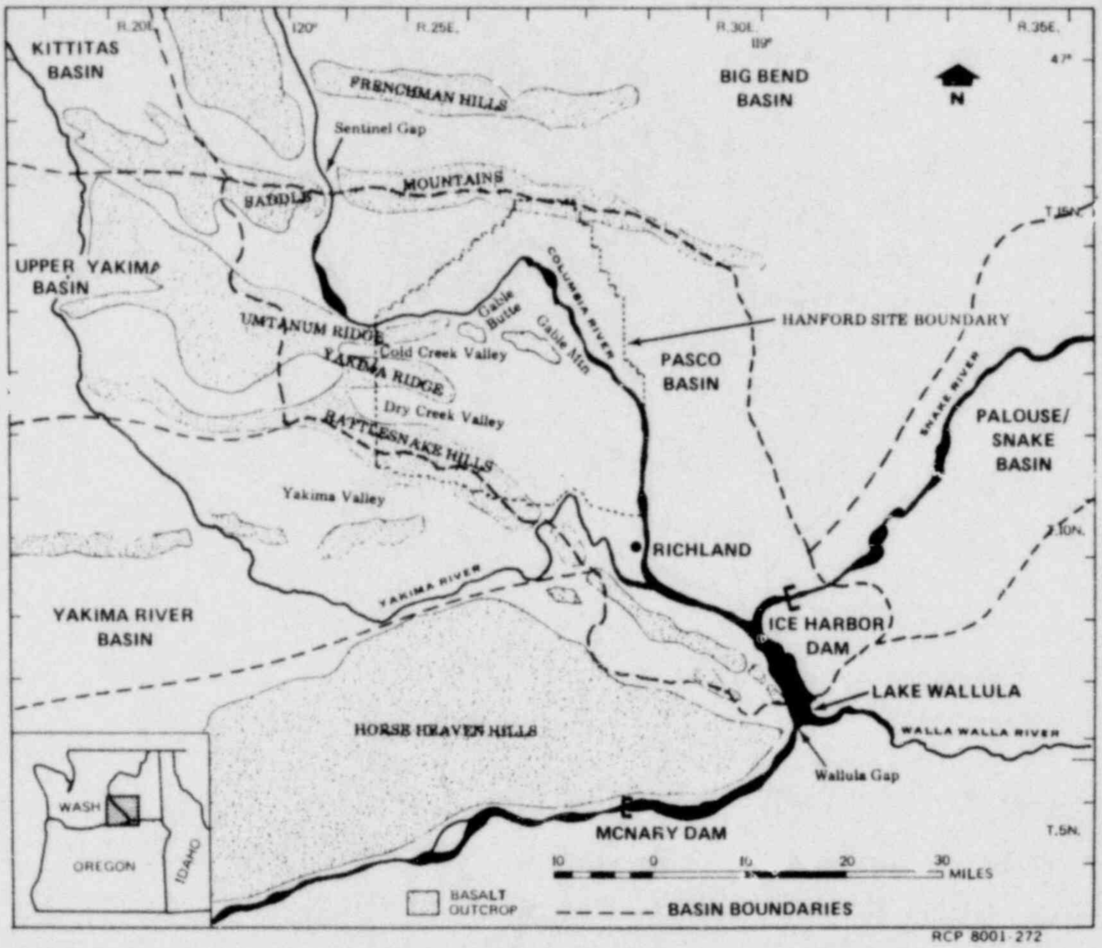


Figure II-18. Location of the Hanford Site in the Pasco Basin

Source: (Reference 289) R.C. Edward, Geophysical Surveys in the Pasco Basin, RHO-BWI-79-100, Rockwell Hanford Operations, Richland, WA, 1979

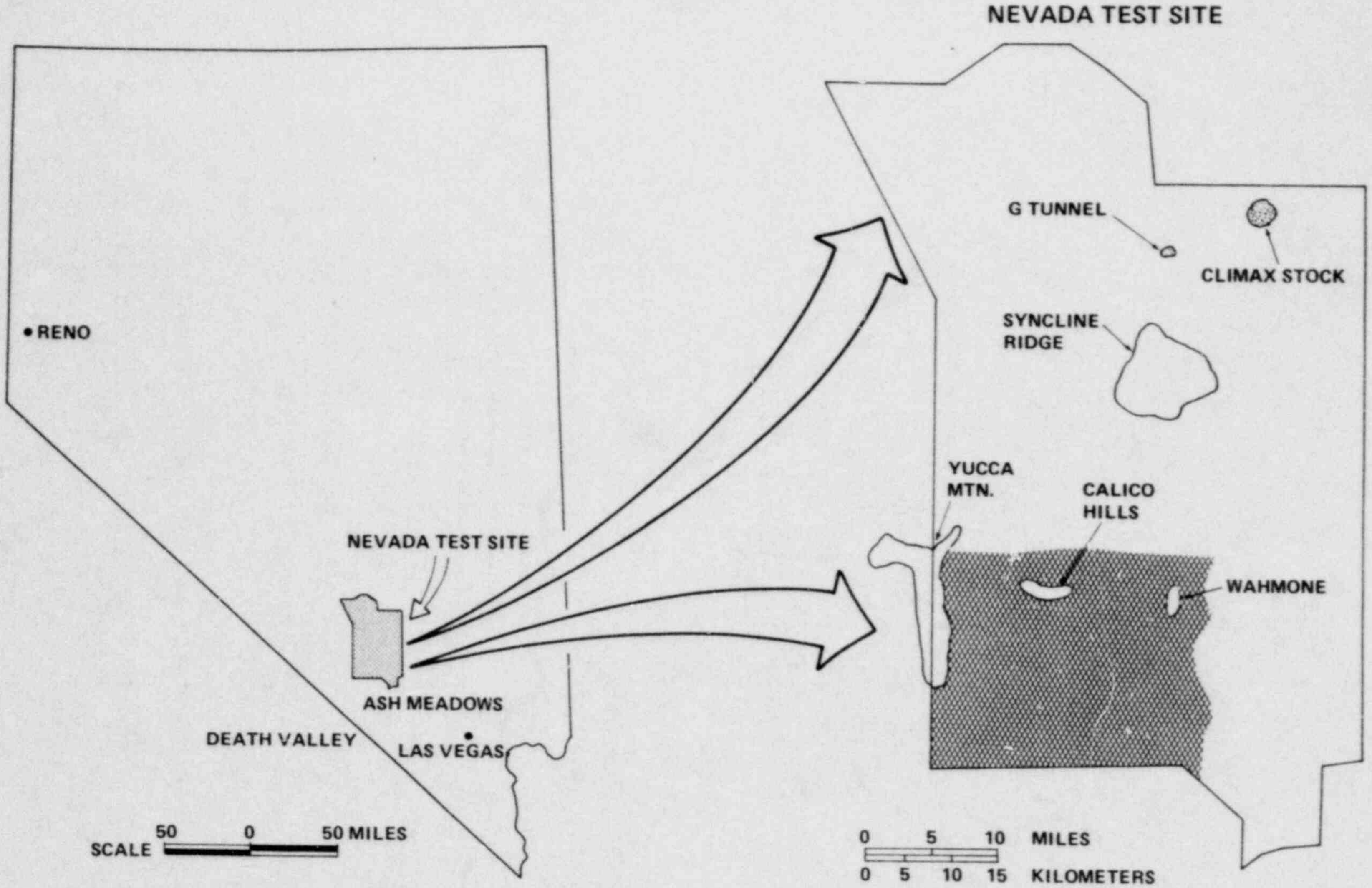


Figure II-19. Location of the Nevada Test Site, The Current Exploration Area (Crosshatched area on right), and Supporting Test Facility Areas (Stippled area on right) portion. The area compatible with weapons testing consists of approximately 300 square miles of desert basins and mountain ranges. Early in the project, sites outside the southwest quadrant were considered. One such site, the Syncline Ridge was eliminated from further consideration because of its structural complexity (295). In addition to exploration for repository sites, field tests are being performed at the Climax Stock and G-Tunnel, to evaluate the suitability of granite and welded tuff, respectively, as host media. A summary of the investigations conducted to date is available in a recent report (296).

Current data and understanding indicate that the Nevada Test Site and tuffaceous rocks have potential for isolating radioactive waste in a mined geologic repository. The identification of a site with suitable geologic and hydrologic characteristics is the primary focus of exploration. The likelihood and the consequences of potential tectonic events are being analyzed to determine whether the tectonic setting would preclude waste disposal at the site. Preliminary data suggest that the risks from tectonic phenomena are acceptable. Metallic and energy-resource conflicts are minimal; both water and land-use resources need careful evaluation.

The geology of the Nevada Test Site is complex, a characteristic shared by all of the Basin and Range Province in which the NTS is located. The geologic strata present at the NTS consist of more than 30,000 ft of pre-Cambrian and Paleozoic quartzites, shales, and carbonates (297). This sequence of sedimentary rocks was deformed during compressional mountain-building episodes in Mesozoic time. During the mid-Cenozoic, a few thousand feet of volcanic deposits were deposited in the region and subsequently displaced by normal faults (297, 298). The closed basins associated with the normal faulting have accumulated alluvium deposits that exceed 2000 ft in thickness in the deepest parts of the basins (297).

Of the rock types that occur the Nevada Test Site, argillite, granite, alluvium, and tuff have been considered for suitability as host rocks. Alluvium was deferred from consideration as a candidate host for high-level waste because its thermal conductivity (assumed for the study to range from 0.2 to 1.2 W/m-K) would allow unacceptable near-canister temperatures for 10-year-old high heat generating wastes (299). Geophysical data collected during 1978 and 1979 indicated the presence of structural discontinuities passing through the Calico Hills (argillite-granite) and Wahmonie (granite) study areas. Magnetic and gravity data suggested a possible granitic intrusion at shallow depth below the argillites at Calico Hills, but a 2,550-ft drill hole failed to penetrate the inferred granitic mass (210). Therefore, current exploration efforts in the southwestern part of the Nevada Test Site are directed to locations containing the remaining candidate host rock, volcanic tuff. At present, only one location, Yucca Mountain (Figure II-19), is being explored.

Yucca Mountain is underlain by approximately 2000 m of interbedded welded to nonwelded tuffs. The thermal, mechanical, and chemical properties of welded tuff are generally considered favorable for a geologic repository (300). The thermal conductivity of saturated welded tuff is about 2.5 W/m-K, and its mechanical strength approaches that of granite (164). However, welded tuff may contain up to 10% water by weight. If this water is removed from the rock, thermal conductivity decreases to approximately 1.7 W/m-K. The effects of this water on the thermomechanical response of tuff have to be assessed and are being investigated by an at-depth heating test at G-tunnel (247) and by laboratory studies and computer modeling. The sorption capacity of tuff is generally very high and quite variable. Depending on the mineralogy, sorption values for various types of tuff range from 55 to 13,000 ml/g for strontium and from 50 to 6,000 ml/g for cesium.

An ideal geologic setting for a repository in tuff is a thermally conductive, mechanically strong, welded tuff enveloped by a low-permeability, highly sorptive, nonwelded zeolitized tuff (Figure II-20). Field mapping, core drilling, and geophysical surveying are in progress to assess the extent to which these conditions exist at Yucca Mountain. A 6,000-ft core and hydrologic test hole is being drilled into the study area; the results will be correlated with data from a 2,500-ft hole drilled earlier (209). The water-bearing properties of inferred fracture zones in the Yucca Mountain area will be evaluated by hydrologic testing and geophysical surveys.

The Nevada Test Site lies in a portion of the Great Basin characterized by large, closed ground water and surface water basins. Ground water flows southwest and discharges at Ash Meadows, Nevada, and Death Valley,

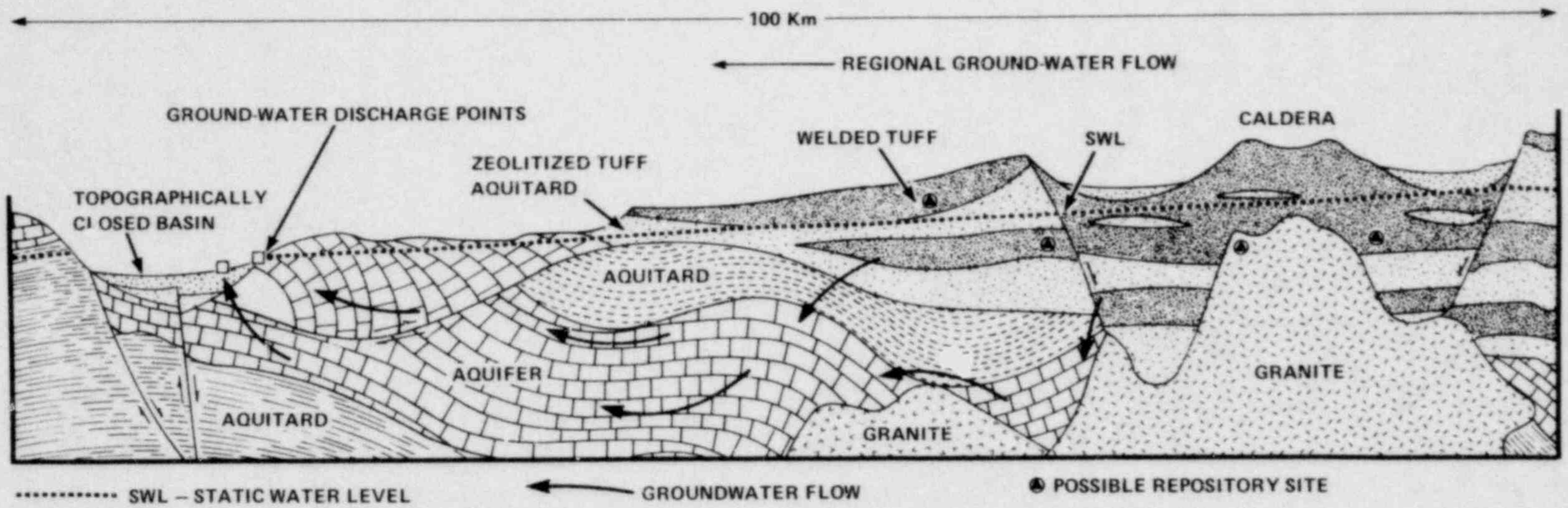


Figure II-20. Idealized Settings for a Repository in Tuff

California (Figure II-19), about 40 and 60 km from the current study area, respectively (238). The water table is between 400 and 600 m deep in the study area (209). Few reliable estimates of ground water flow velocity are available for the NTS region. Estimates based on radiocarbon ages of water from wells at the NTS and from wells in the Amargosa Farms area, approximately 32 km downgradient, indicate an average velocity of about 9 m/yr between these wells (296). This rate is probably higher than average ground water velocities in the NTS because much of the flow to the wells is through local zones of fractured tuffaceous rocks and alluvium whose average hydraulic conductivity is higher than for other areas at the site with the exception of some areas of carbonates (238). A two-dimensional, finite-difference model of the regional flow system encompassing the Nevada Test Site is currently being tested by sensitivity analyses to determine the relative importance of the model's hydrologic parameters to velocity and residence-time calculations (301).

The Nevada Test Site is in an area of the Great Basin that has relatively low seismic activity for the Basin and Range Province (seismic risk zone 2) (177, 261, 302) and a low potential for volcanic eruptions (181, 183). Based on the historic seismic record within 400 km of the NTS, it is estimated that an acceleration of 0.7g has a return period of 15,000 years, and 0.5g has a return period of about 2,500 years (261). Methods for estimating the recurrence potential of basaltic volcanism yield preliminary annual probabilities for a 10 km^2 area of 10^{-8} to 10^{-9} per square kilometer (183). The calculations assume that volcanism is a temporally and spatially random process within the NTS region. This assumption is conservative. Current work, including field and geophysical studies, is concerned with identifying structural features that affect the distribution of past volcanic activity. The relationship between Quaternary strata of the basin fill deposits and erosional and depositional surfaces indicates that only very slight regional and local uplift and subsidence have occurred during the past few million years (303).

Limited mining was conducted in the area of the Nevada Test Site long before the land was withdrawn. The mineral deposits are generally associated with intrusive granitic masses. All potential repository sites on or near the site will be evaluated for mineral resources. No hydrocarbon deposits of reasonably foreseeable commercial grades occur at the Nevada Test Site. However, because the NTS is in a desert, water resources are scarce and must be considered.

II.D.5.7 Expanded National Waste Terminal Storage Program

The Department's site exploration program is being expanded to consider a wider variety of rock types in diverse geologic environments. These broadened activities were originally recommended by the Interagency Review Group (204) and were included in the President's statement of 12 February 1980 (1).

Three approaches to site exploration and characterization are currently available for use to implement the expanded program at the national screening phase. The approaches are (i) geologic-host rock; (ii) current land use; and (iii) geohydrologic environment. The three site exploration and characterization approaches are described in Section III.C.1.

The geologic, or host-rock, approach has been applied in separate literature surveys of granitic intrusive rocks in the conterminous United States and in the southern Piedmont to determine their distribution and potential suitability as repository host rocks (141, 142). The factors evaluated included physical, chemical, hydrologic, tectonic, seismic, and mineralogic properties. The information collected in these studies indicates that numerous granitic bodies are suitable for further evaluation.

Three evaluations have been completed in the following subregions of the southeastern United States:

1. The Piedmont Province, consisting of pre-Cambrian and Paleozoic metamorphic and igneous rocks (142).

2. Triassic basins, long narrow troughs in the Piedmont of Triassic sedimentary and extrusive rocks (133).

3. The Coastal Plain, a wedge of unconsolidated and semiconsolidated sands, clays, and limestones overlying rocks of the Piedmont type (304, 305).

A literature study of argillaceous rocks in the United States is currently under way. Laterally persistent argillaceous rock units at least 75 m thick and 300 to 1000 m deep are being evaluated. Characteristics being considered include physical properties, mineral composition, geochemistry, geologic structure, seismic potential and tectonic history, development of mineral resources, regional ground water hydrology, and the extent of drilling and mining. This study is expected to be completed in the third quarter of FY 1980.

Three broad classes of argillaceous rocks are being evaluated by laboratory testing:

1. Argillaceous rocks with no carbonaceous material and with little or no hydrous expandable clay.

2. Argillaceous rocks with carbonaceous material.

3. Argillaceous rocks with smectite as a major clay mineral constituent.

Each class will be sampled by drilling and evaluated to determine the major mineralogic constituents. Each sample will be characterized by (i) complete chemical and mineralogical analyses; (ii) response or reaction to large doses of gamma radiation; (iii) measurement of the thermal properties; (iv) measurement of absorptive properties; (v) measurement of the mechanical properties; and (vi) determination of the products of the interaction between the samples and simulated radioactive waste. This project will continue through FY 1981.

Screening based upon consideration of geohydrologic environments screening is being initiated to evaluate simultaneously the literature regarding geohydrologic and environmental requirements. This study will entail a nationwide literature search and result in a computerized data base suitable for use in parametric analyses. Concurrently, the U.S. Geological Survey is planning evaluations of various geohydrologic provinces of the United States.

II.D.6 Natural Systems Summary

The information presented in Sections II.D.4, II.D.5, and Appendix B indicates the technical scope and current status of the Department's programs to evaluate the potential of various natural systems to provide the necessary features for a repository. The natural and engineered barrier systems discussed in the next chapter are closely related because the construction and operation of the repository will affect the natural systems. If appropriately designed, engineered systems can supplement the containment and isolation capabilities of the natural system. Much of the information gathered during field and laboratory investigations of natural systems will be used in the modeling programs described in Chapter II.F.

The information about the natural system presented in this chapter and Appendix B can be summarized as follows:

1. The scope of technical information required for evaluating natural systems and the role that natural systems can play in providing barriers for containment and isolation are known.

2. Required characterization techniques are available; many represent the state of the art.

3. The need for additional improvement in predicting the performance of fractured, and perhaps water-bearing, rock masses has been recognized.

4. Site identification programs are being conducted in a number of regions and host rocks, including basalt, granite, shale, salt, and tuff; some are well advanced. A geologic characterization report and a draft environmental impact statement have been prepared for a bedded-salt site in southeastern New Mexico from data gathered in extensive site drilling and field characterization.

   Preliminary drilling and other field activities are being conducted at potential locations (see III.C) on the Department's Nevada Test Site and the Hanford Site in volcanic tuff and basalt environments, respectively. Eight salt domes in the Gulf interior region were identified for area characterization studies; the Department is continuing evaluation of seven of these for possible location characterization studies.

Area-level studies (see III.C) are also currently being conducted in the Paradox basin of Utah; the Appalachian basin of New York, Ohio, and Pennsylvania; and the Palo Duro and Dalhart basins in Texas. Regional level studies (see III.C) for granite and shale have been completed, and area studies will soon commence.

5. Investigations to date strongly suggest that acceptable natural systems exist that will meet the objectives in II.A.1.

6. The diversity of media under evaluation, the large number of potentially suitable sites contained in the areas and regions being studied, and the NWTS Program's ability to successfully screen for sites using criteria (II.D.3) and the available performance assessment techniques (II.F.1) will result in identifying, qualifying, and licensing repository sites.

II.E MAN-MADE SYSTEMS OF MINED GEOLOGIC DISPOSAL

Chapter II.D describes the natural systems associated with mined geologic disposal. This chapter addresses the subject of man-made systems. The man-made repository systems consist of three basic functional components: the waste package system (II.E.1), the repository system (II.E.2), and preventive measures against human intrusion (II.E.3). This chapter addresses the factors requiring consideration for each of these man-made system components and presents (i) the phenomena of concern, (ii) a description of and the requirements for successful functioning of the components, and (iii) the status of knowledge and planned activities relating to each.

II.E.1 Waste Package System

The waste package, as used here, includes everything man places in the repository waste emplacement hole, e.g., the waste form, canister, overpack, and backfill. These various package system components will be used to reduce overall technical uncertainties by virtue of their conservative engineering design and by providing attenuation of risk to man in addition to that provided by the host rock and surrounding strata. These components provide:

1. Containment of fission products for extended periods to allow short-lived radionuclides to decay while still localized.

2. Limitation of the rate of release of radionuclides to the near-field host rock for the entire period of isolation.

3. Limitation of access of water to components of the packages to prevent or minimize waste/rock/leachant interactions.

The functions and materials of package system components will vary in response to specific site needs and environmental factors. A waste package system is conceptually represented in Figure II-21. The package contains a number of components which might be appropriate under particular circumstances, but not all components are necessarily appropriate or required for a specific system. The choice of waste package components will depend on the characteristics of the waste to be contained and the characteristics of the natural geologic system.

One may envision how this multibarrier package system performs by considering the case of ground water moving though the natural system toward the package. Water first would encounter the emplacement hole backfill, which could be designed to be relatively impermeable to water; such design would make use of well-established physical and chemical reactions.

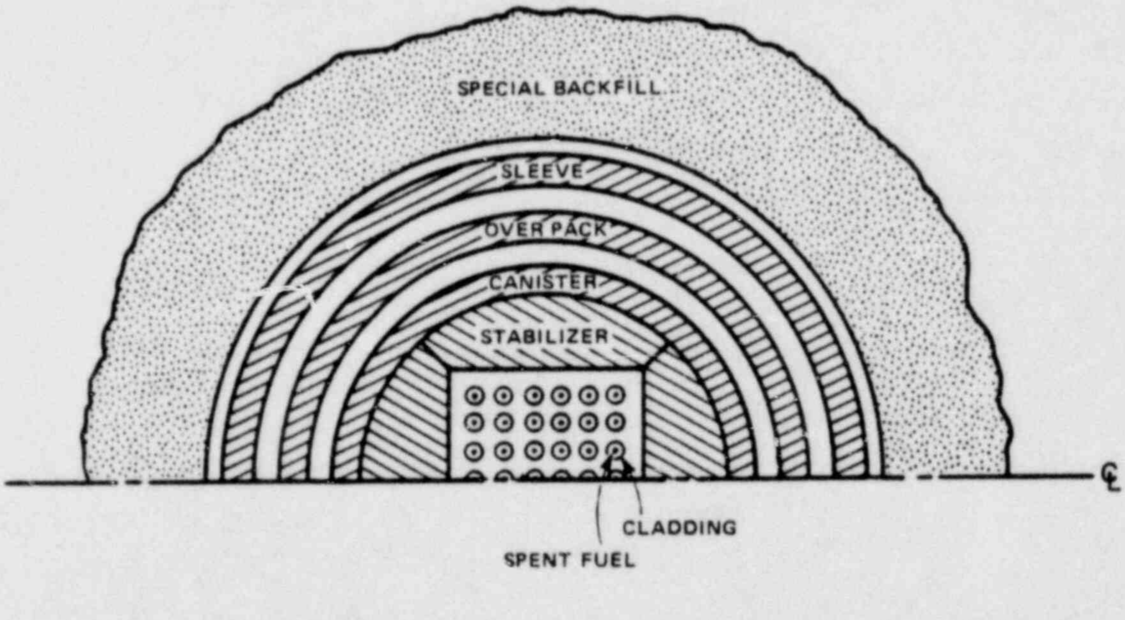


Figure II-21. Conceptual Waste Package

Any residual water passing through the backfill would encounter a sleeve, an overpack, or both, made of corrosion-resistant materials. As a further redundant measure, the canister itself would act as a physical barrier. Use of a nongaseous stabilizer could provide an additional resistance to water influx. If all these sequential resistances to water influx were to fail, the waste form itself would be a barrier because of its resistance to radionuclide release. If some nuclides were mobilized by the ground water, they would have to travel through package components. The backfill or other parts of the system could then function as a sorptive barrier to retard or minimize movement of selected nuclides. Thus, the total waste package system can be designed to minimize the nuclide inventory entering the natural system. Chemically and physically limiting nuclide mobility, and also delaying nuclide releases, results in a substantial decay before nuclides enter the natural system, where there would be additional delay and decay due to sorption reactions with surrounding rock materials along the ground water path.

The candidate package components are discussed subsequently in more detail relative to the phenomena and environments (see also II.C) to which the package may be exposed. The functions of package components when exposed to those conditions, the expected requirements that will be placed on the performance of the waste package system, and the current state of our knowledge are described.

II.E.1.1 Phenomena of Concern

Numerous phenomena that initiate reactions in and stresses on the waste package system have been identified. These phenomena occur both in the operating phase of the repository, e.g., handling and emplacement processes, and during the long-term isolation phase of the repository, e.g., barrier deterioration as a result of corrosion or chemical and radiolytic degradation. Operating phase considerations are discussed in II.F.3. The principal phenomena of concern to the long-term isolation phase are as follows:

1. Incursion of fluid due to natural or anthropogenic effects.

2. Volatilization of nuclides.

3. Radiolysis.

4. Heat generation and thermal-mechanical stresses on package components and host rock.

5. Interactions with ground water.

6. Other interactions among package components and host rock.

The processes to be considered depend upon the emplacement medium and the temperatures and pressures that the waste package system must withstand. The package thus must be designed to confine the waste under widely varying conditions.

The waste package will consist of various components, each of which can mitigate the impacts of these phenomena on package performance. It is important to note that the design may require more than one component to resist all potential phenomena. The use of a multicomponent waste package design will increase the ability of the package system to resist all expected phenomena.

II.E.1.2 Description of Package Components

The waste package has been defined as a system of engineered components that may include waste form, stabilizer, canister, overpack, sleeve, and emplacement hole backfill (306). These components may be exposed to a variety of phenomena that can be defined in terms of anticipated repository environments (II.C). The following paragraphs discuss each package component separately; however, it is the performance of the entire system of components on which containment of the waste depends.

II.E.1.2.1 Waste Form

The waste form in the reference case for this Statement is spent fuel as discharged from light water reactors (LWR's) with little or perhaps no modification other than aging or possible mechanical disassembly to allow volume reduction (see I.A). The waste form function is to provide a durable barrier to delay or minimize radionuclide release. Since the most credible transport mechanism to the biosphere is by ground water movement, the leaching or dissolution (release) characteristics of the waste form are of concern. Preserving the integrity of the waste form would minimize the mobilization of the radionuclides. For spent fuel, the properties of the cladding must also be considered because it may provide an additional barrier against water exposure of the fuel. Also, the cladding contains fission product gases released from the spent fuel, e.g., tritium, xenon, and iodine, and prevents their entry into free spaces within the canister. Other properties of the waste form that must be considered include thermal/physical response, mechanical shock resistance, phase stability, volatility and gas generation, and compatibility with other components within the waste package system.

II.E.1.2.2 Stabilizer

The stabilizer (sometimes referred to as filler) can have several functions within a spent fuel container. Gases, such as helium, could enhance heat transfer during the thermal period without causing chemical or mechanical attack on the spent fuel/cladding assembly or the canister. Particulate or solid stabilizers can provide additional functions, including maintaining the position of the spent fuel within the canister; preventing canister collapse under lithostatic pressures; and acting as a corrosion-resistant protective barrier, improving heat transfer, increasing radiation attenuation, retarding water intrusion, and enhancing desiccant activity or nuclide sorption.

II.E.1.2.3 Canister

The canister provides containment for the waste forms and stabilizer. It should delay or minimize waste/water interactions as long as practicably achievable. Moreover, the canister is expected to provide physical protection during interim storage, transportation, handling, emplacement, and, possibly, waste retrieval operations.

II.E.1.2.4 Overpack

The overpack is similar in principle to the canister. An overpack offers several options to the package designer: it may function as a redundant canister, applied (if necessary) for all stages of package handling, transportation, and emplacement; it can exhibit corrosion resistance or mechanical properties superior to those of the original canister, thereby providing all, or a major part, of the resistance to the environment required by the package longevity criteria; it can provide a degree of uniformity to a variety of canister types to accommodate repository acceptance criteria. The canister and overpack together can be referred to as the "container."

II.E.1.2.5 Emplacement Hole Sleeve

The function of the emplacement hole sleeve, for current waste package concepts, is to maintain an open emplacement hole for easy package insertion and retrieval. This may be important if the geologic medium is plastic, e.g., rock salt and certain shales. However, the sleeve also could assume a more significant package role. For example, certain barriers could more easily be formed or built in situ rather than transported and emplaced with the canister, e.g., poured concrete sleeves. Such sleeves could function as barriers to the entry of ground water.

II.E.1.2.6 Emplacement Hole Backfill

Emplacement hole backfill materials which will be used to fill the void spaces between the canister, overpack, sleeve and host rock, may have several functions:

1. Sorbing the limited amount of water present in a dry repository medium, e.g., brine inclusions in salt.

2. Impeding movement of ground water to or from the waste package.

3. Selectively sorbing radioisotopes from the water that contacted the waste in the event of a canister breach.

4. Modifying the ground water chemistry and composition (pH, Eh, selectively [illegible]d chemical species) in the vicinity of the package to reduce corrosion rates or minimize waste form leaching.

5. Providing mechanical relief for accommodating hole closure and stresses on the waste package induced by rock movement.

6. Serving as a heat transfer medium.

II.E.1.3 <u>Waste Package Requirements</u>

The intention of waste package performance requirements is to ensure that the waste package design will conservatively satisfy overall geologic disposal system requirements and that it will provide the functions required for handling, emplacement, and, if necessary, retrieval of the waste. The requirements apply to disposal in any geologic medium.

The waste package or, in some cases, specific components of the package will be required to perform three functions within the repository system:

1. The waste package must act as a barrier to radionuclide mobilization and release into the geologic system over long periods of time in accordance with Objective 1 stated in II.A.1.

2. The waste package must provide for the safe handling of waste at the repository in conjunction with other components of the repository system.

3. The waste package must preserve the ability to retrieve the waste safely throughout the required repository demonstration period.

These capabilities will be required under both normal operating conditions and specified design basis accident conditions to ensure that, in conjunction with other components of the repository system, the performance of the repository is in compliance with overall system requirements.

Waste handling operations at the repository will include receiving, inspection, transfer to the underground, and emplacement. The capability to package or overpack spent fuel on-site may be included. In this case, any receiving and inspection requirements would apply to the spent fuel shipping cask as well as the final spent fuel container used for disposal.

The safety provided by the containerized waste during handling and emplacement operations will be assessed on the basis of its ability to do the following:

1. Maintain waste containment under normal operating stresses due to vibration or impact.

2. Limit radionuclide dispersion under accident conditions involving combustion.

3. Maintain surface contamination within the limits specified for repository handling components.

4. Limit hazards to the public and the repository personnel from exposure to chemically toxic substances.

5. Prevent criticality within single packages or an array of packages by virtue of the physical properties and the spatial arrangment of a group of packages.

The conditions under which the package must perform will range from normal operating conditions to abnormal conditions due to minor operational incidents or to conditions resulting from design-basis accidents. Under the normal and abnormal operating conditions, the waste container will retain its physical integrity and be capable of providing containment throughout the handling and emplacement operations. After more severe accident conditions, repackaging or overpacking the damaged container will be performed if required.

The waste container and overpack design will be standarized to the extent practicable. Standardization is primarily for operational uniformity and does not apply to material types chosen on a site-specific basis. The features of the waste package to be considered for standardization include package dimensions, geometry, maximum weight, and allowable package surface contamination. The standardization of the package components not only represents good engineering practice but also contributes to safety by ensuring that wastes can be handled at any repository, using existing handling facilities at the repository, and that surface dose rates will be within design limits for all repository handling facilities.

If for any reason a repository is found to be unacceptable during the operating period, the waste will have to be retrievable (II.F.3). Retrievability requires that the containerized waste remain intact throughout the operating period and this imposes durability requirements on the containerized waste.

In addition, the waste package will provide a long-term barrier to radionuclide release and transport into the surrounding rock strata. It is important to understand that the package will be evaluated as a system. Under certain conditions, some components of the system may not be entirely effective; thus the burden of some portion of their function falls on other components. The package system is being technically evaluated to ensure that, if individual components fail, the package will maintain its ability to act as an effective deterrent to radionuclide transport.

II.E.1.4 Current State of Knowledge Regarding Waste Package System

Extensive testing and development studies on various individual barrier components of the waste package system, under expected conditions of geologic isolation (II.C.4), have been in progress for several years. These studies have been conducted in industrial and national laboratories, as well as universities, both in this country and abroad. Most of these studies are not complete, but the data and results generated during the past few years do indicate that components of the waste package system can prevent or minimize release of radionuclides to the natural system by functioning as effective chemical and physical barriers. Programs, program plans, and results are described here.

Because of the many candidate materials suggested for each of the components of the waste package, barrier development programs are proceeding in a logical sequence of scale and complexity of the tests. Generally, studies are started in the laboratory, usually on single components, with tests directed toward the major properties of interest associated with the prime function of the barrier. The initial laboratory tests in the Department's program utilized simulated waste forms without the influence of a radiation field. Later, laboratory studies started to utilize real waste (307-310). Experiments that simulate repository conditions and integrate the behavior of waste package components and the geological media are in progress (311, 312). Further laboratory efforts (306, 313, 314) are being directed toward more complex testing of subsystems involving more than one component.

Through laboratory materials performance evaluation under realistic repository environmental conditions and accelerated aging tests (referred to as "overtests"), the number of waste package candidate materials is being reduced for subsequent evaluation (315).

Following laboratory testing, cold "bench-scale" experiments and hot-cell experiments are planned (313); some are already under way. These tests employ small-scale mockups of real, complex systems or groups of system components to investigate the influence of components upon each other. For example, leaching/corrosion studies utilizing a scaled mini-canister of an actual waste form with rocks and ground waters are in progress (316, 317).

The logical culmination of a series of studies investigating waste package component performance and qualification is a large-scale test specific to each repository rock type (318, 319) (see also II.D.1 and II.F.2), which involves all components of the waste package. Such tests also may be performed in the field to confirm the results of earlier detailed laboratory and large-scale tests. To evaluate the response of the total package system it may be necessary to induce failure in one or more component in a realistic way, so as to stress one or more of the remaining components. If, for example, the backfill successfully excludes water from the overpack, no information will be obtained about the in situ resistance of the overpack to the stress of ground water. The required extent of such field testing will be determined from the analysis of earlier results.

Various aspects of required laboratory, large-scale, and field tests have been described by the U.S. Geological Survey and the Department in the Earth Science Technical Plan (314) which discusses the types of data required and the sequence of laboratory, large-scale engineering, and field demonstration tests. A Waste Package Design, Development, and Test Plan (306) has been formulated to direct these package development efforts. An integral part of this plan is the development of coordination among and standards to be followed by researchers and waste management program entities with respect to testing procedures and materials certification. For review and integration, the Department has established a Materials Steering Committee, a Materials Review Board, a Materials Characterization Center, and an Independent Measurement Standards Laboratory (320). These organizations have been charged with supporting waste package design, development, and testing programs to produce suitable packages that meet established requirements (see also Part III).

II.E.1.5 Status of Package Components

II.E.1.5.1 Waste Form

Spent fuel as a waste form, has a preset form, composition, and geometry (321). The in-reactor operating history (322, 323) and also its postoperating and storage history will dictate the physical condition of the spent fuel as received for packaging and disposal (see II.C). The actual physical condition will influence the containment efficiency of the cladding, the release resistance of the fuel pellets, and the radiation and thermal output of the packaged spent fuel (307, 310, 324). The majority of the spent fuel assemblies will be of the PWR and BWR types consisting of spent fuel pins of uranium dioxide pellets in sealed metal (Zircaloy) tubes held in a geometric space array by a metal framework. Concepts to prepare spent fuel for disposal now under investigation include the following, listed in order of increasing complexity (325-327):

1. Intact fuel assembly: minimizes handling and packaging operation and also the need for additional waste streams; is compatible with storage concept.

2. Removal of assembly nozzles: creates additional waste stream and packaging operation but permits shorter package length; is a necessary step for other concepts.

3. Fission gas removal and reseal: has same attributes as removal of assembly nozzles and also prolongs cladding integrity and reduces canister pressurization and subsequent release if breached.

4. Fuel bundle disassembly: has same attributes as removal of assembly nozzles and also eliminates effect of dissimilar metal contact on cladding corrosion; may lead to possible temperature effects due to close packing of pins.

5. Chopped fuel and matrix immobilization: has attributes of fission gas removal and reseal although it destroys cladding as containment, but provides uniformity of waste form and new barrier option, i.e., matrix material.

Spent fuel as a waste form has been under investigation for some time (319, 325, 326, 328, 329). Laboratory studies on the release of radionuclides from spent fuel (307-310, 330) have included leachants of various ground water composition. These tests are continuing, along with new studies to strengthen the understanding of the effects of degree of oxidation and chemical distribution on nuclide release from spent fuel (331). These tests are being performed under both oxidizing and anoxic environments in various ground waters.

A naturally occurring deposit of reactor wastes was created 2 billion years ago at the site of a current uranium mine (Oklo) in Gabon, West Africa (332). Certain regions accreted enough very rich ore to become "critical" and sustain a fission chain reaction for several hundred thousand years. More than 10 tons of wastes, comparable in nearly every respect to the products generated by a modern power reactor, were formed in the buried ore where they have been exposed to dispersive natural processes ever since. Yet the majority of the waste products are still at or near the original site.

Better understanding of the Oklo phenomenon will increase our knowledge of how a number of waste products in spent fuel will behave over extremely long periods of time. To date, studies show that the long-term release rate of most of the Oklo reactor products is controlled by solid-state diffusion from crystalline grains of uraninite, a process which is very slow compared to processes which involve direct dissolution of the ore. At Oklo the uraninite grains remained undissolved because oxygen, which is necessary to oxidize the uranium to a soluble form, was held to extremely low values in the surrounding ground water by the presence of an excess of reducing materials.

In order to determine whether present-day spent fuel can be expected to behave similarly in a similar geochemical environment, studies are being conducted to determine whether the release rates of waste nuclides are controlled by diffusion from UO_2 when the oxygen content of water is held to very low values (330). To date the information obtained from such experiments indicates that lowering the oxygen content of the water can significantly decrease the release rate of the wastes.

Accelerated release studies are currently investigating the behavior of a simulated spent fuel in both deionized water and a simulated salt brine under hydrothermal conditions of $100^{o}C$, $200^{o}C$, and $300^{o}C$ at pressures up to 300 bars (333-335). The alteration and degradation of the original solid and the chemical species taken into solution are being studied.

Preliminary results from these various studies indicate that, although some radionuclides are released more rapidly than others as a function of experimental conditions, spent fuel is a durable waste form that exhibits low release of radionuclides when subjected to ground water under normal repository conditions.

II.E.1.5.2 Stabilizers

For stabilizers (fillers) in spent fuel canisters, candidate materials include inert gases (e.g., helium) and castable solids (e.g., glass, lead and lead alloys, zinc and zinc alloys) (329, 336). A list of candidate canister stabilizer materials follows (337):

Gases:

Helium
Nitrogen

Solids:

Fe, Fe alloys	Graphite
Pb, Pb alloys	Bentonite
Zn, Zn alloys	Other clay minerals
Glass	Sand
Concrete	Crushed host rock
Al, Al alloys	Mixtures of the above
Cu, Cu alloys	

Basic physical and chemical properties of candidate stabilizer materials are well known. Some of these candidate materials have been evaluated (under expected repository conditions) for use as barrier materials other than as stabilizers, e.g., as canister, overpack, and/or backfill barriers. Since the overall waste package functions are similar (e.g., corrosion resistance, nuclide sorptive properties, protection of the waste form), the same materials testing can, in many cases, be applied to several system components. Results of such testing (315, 337-341) for metallic, ceramic, and other materials are discussed below.

II.E.1.5.3 Canister, Overpack, and Sleeve

Candidate materials for the canister, overpack, and sleeve can be discussed together, since they all are basically impermeable or "hard" package elements and will act in much the same manner though perhaps for different package functions. Candidates include metals, ceramics, carbon, glasses, and cements. A selected list of candidate materials follows, with each of the major materials groups discussed in subsequent paragraphs:

Metals:

Ti alloys
Zr alloys
Ni alloys

Pb, Pb alloys
Fe, Fe alloys
Cu, Cu alloys
Aluminum alloys

Ceramics:

Al_2O_3 (alumina)
$2Al_2O_3 \cdot SiO_2$ (mullite)
Al_2O_3-ZrO_2-SiO_2 (fused AZS refractory)
$CaTiO_3$ (perovskite)
$CaO \cdot ZrO_2 \cdot TiO_2$ (zirconolite)
TiO_2 (rutile)
ZrO_2 (zirconia)
$ZrSiO_4$ (zircon)

Carbides:

TiC
SiC
TaC

Carbon:

Impervious graphite
Glassy carbon
Pyrolytic graphite

Glasses:

Wide variety

Cements:

High-alumina cements
Specially tailored cements, grouting
Compounds, or chemical binders

Candidate material selection for canister and overpack will be based largely on the results of corrosion tests as a function of temperature, radiation, and ground water chemistry, e.g., pH, Eh, composition, and ionic strength, that are typical of the water in various media of interest, i.e., basalt, granite, salt, and tuff. Applicable materials studies to date include consideration of general corrosion rates, pitting and crevice corrosion susceptibilities, stress corrosion cracking, effects of oxygen concentration, solution volume to solid surface area ratio, and possible effects from radiolysis products (315, 338, 342).

II.E.1.5.3.1 Metals

More data exist on the behavior and corrosion resistance of metals under anticipated repository environments than any of the other listed groups. The specific nature of the environments involved (II.C), i.e., brines, bitterns, or hard rock ground water, can lead to markedly different material requirements and characterization studies (337).

Several studies (315, 338, 342, 343, 344) have reported data on the corrosion of candidate metals in brines under both oxygenated and anoxic conditions. These data, obtained for temperatures up to 250°C, are applicable to bedded and domed salt waste repositories as well as subseabed disposal (315). Corrosion data for candidate metals (315, 338, 342) under representative as well as severe overtest brine conditions demonstrate the long-term durability of metallic barriers. Certain titanium and nickel-based alloys show outstanding general corrosion resistance (315) to high-temperature brines (70°C to 250°C) under both anoxic and oxygenated conditions, even in the presence of a radiation field (338, 342).

Based upon laboratory test data, thin layers of these nickel or titanium materials, 1/16 to 1/4 in. thick, should remain intact for hundreds of years or more under these hostile conditions. Further studies in brine are continuing (338) so that localized attack (e.g., pitting, crevice corrosion, stress corrosion cracking) can be more fully characterized under the same conditions. Observed localized attack on several of the most promising candidates has been minimal (315, 338, 342).

Objectives of continuing metallurgical compatibility testing include (315) better definition of reaction mechanisms and kinetics, additional data necessary for repository design and performance assessment analysis, and developing and testing of predictive analytical models.

Corrosion studies relevant to high-temperature geothermal brine applications (343, 344) performed under anoxic conditions are in good agreement with the geologic repository salt brine investigations (315, 342). In both cases, titanium alloys show outstanding performance as a barrier (i.e., canister, overpack, sleeve) material, and several nickel-based alloys show comparable properties.

Metal corrosion data relevant to materials selection for hard-rock repositories (e.g. basalt, granite, tuff) under expected repository conditions are not as extensive, but additional studies are planned (306). Hard rock ground water is not expected to be as corrosive as brines, however, and there are applicable results from the Swedish waste management program (345-347). For unreprocessed spent fuel disposal in granite, thick copper canisters were evaluated to have service life in ground water of thousands to hundreds of thousands of years.

Although evaluation of candidate metallic barrier materials is not complete, available data indicate that several alloys are capable of maintaining their barrier integrity for extended time periods under expected salt or granite repository environmental conditions. Metallic barriers are therefore expected to have a significant role in spent fuel package systems.

II.E.1.5.3.2 Ceramics

In general, ceramics offer good resistance to chemical degradation. Several candidates have been selected which show strong resistance to phase changes and hydration reactions (337). Also, in general, ceramic materials are very resistant to radiation damage (348). Additional work on these materials as package components is planned (306). As part of the Swedish waste disposal program, aluminum oxide canisters formed by hot isostatic pressing have been investigated for spent fuel disposal (347, 349). Aluminum oxide appears in nature as corundum or sapphire. Like the diamond, corundum is among the hardest natural materials known offering very high chemical resistance over long (geological) periods of time (349). Tests in granitic ground waters are being conducted at temperatures from 100°C to

350°C. Preliminary results show that a canister of aluminum oxide with 100-mm-thick walls can conservatively withstand the action of the ground water for thousands of years (347).

II.E.1.5.3.3 Carbon

Carbon in its various forms offers an excellent combination of properties as a barrier to long-lived radionuclide transport. It is extremely inert to essentially all reactive species at the temperature anticipated in geologic repositories (337). Its stability in high-temperature radiation fields in nuclear reactors has been extensively investigated and verified (350). It is relatively inexpensive and easy to fabricate. Pyrolitic carbon has been investigated as a coating for HLW pellets from reprocessing (351) to increase their leach resistance but has not yet been specifically investigated as a component in a spent fuel package.

II.E.1.5.3.4 Glasses

Many glasses and glass-ceramic materials may be suitable as candidates for waste barriers in the form of canisters, overpacks, or hole sleeves. In general, glasses offer easier fabricability than ceramics, with perhaps slightly less desirable chemical and physical properties when compared to the most stable ceramic forms (337). Although easily discolored, glasses, like ceramics, are very radiation resistant materials (352-357). Because of their chemical stability (leach resistance and low solubility) as well as radiation and thermal stability, glasses have been extensively investigated as a waste form material (358-364). Natural glasses are well known and recognized to have weathered well (365). Also, a glass-ceramic material, Corning's spodumene glassceramic Code 9617, has received preliminary evaluation as a canister material for disposal of spent fuel in a granite environment (346, 349); these early results indicate a corrosion rate of only about 0.01 cm/1,000 yr.

II.E.1.5.3.5 Concrete

Concrete or grout cast in place to form a massive barrier appears promising in sleeve applications. Work on cementatious borehole plug materials shows those materials have very low permeabilities (366). Used as monolithic concrete hole sleeve, such materials would be inexpensive and extremely resistant to water ingress. Ordinary hydration-type concretes may be limited unless shielding by other package components can be used to mitigate radiolysis effects. Also temperature levels must be kept below the maximum tolerated by a given concrete type. However, as noted above, work with certain concrete material has shown promise (367-371).

II.E.1.5.4 Emplacement Hole Backfill

Studies and performance testing on emplacement hole backfill materials have been in progress for several years, in both the United States and Sweden (339, 340, 345, 346, 372, 373, 374). The following list shows materials which have received attention:

Sand	Attapulgite
Bentonite	Peat
Borates	Other clay minerals
Zeolites	Gypsum
Iron	Al_2O_3
CaO	Carbon
MgO	$CaCl_2$
Tachyhydrite	Sand
Anhydrite	Crushed host rock
Apatite	Mixtures of the above

Backfill materials are being tested for selective nuclide sorption capacities (for fission products and actinides), to eliminate or significantly reduce radionuclide migration through the backfill barriers.* The

*Such materials are sometimes referred to as "getters" due to their ability to retard the movement of certain materials.

capability to prevent or delay ground water flow through the backfill is also being evaluated. Other properties of interest being evaluated (339, 340, 372) are thermal conductivity, mechanical support strength, swelling, plastic flow, and forms and methods for emplacements.

Recent studies (339, 372) have focused on the testing and development of smectite clay and sand barriers in the presence of several brine compositions for utilization in a salt repository. Screening studies have been completed on various clays, natural soils, and zeolites, as a function of solution salinity, pH, flow rate, and temperature. The sorptive capacities for Eu, Pu, Am, Cs, and Sr in brine have been measured for smectite clay (hectorite and bentonite/montmorillonite) and sand mixtures. The actinides plutonium and americium sorb very well (K_d values about 2000 ml/g) and Eu, Cs, and Sr sorb moderately (K_d values about 200 ml/g). Mechanisms of nuclide retention are ion exchange and precipitation, plus other chemical and physical sorptive processes. Recently published data (375) indicate that reaction kinetics are influenced by streaming potentials, chemical speciation, and adsorption selectivity under repository conditions. Existing data (339, 372, 374) suggest that a properly chosen, 1-ft-thick backfill barrier surrounding a waste container could delay the breakthrough of Pu and Am (defined as 1% of original concentration) through the barrier for periods of 10,000-100,000 years, dependent upon the interstitial brine flow rate. Concurrently, the breakthrough of Cs, Sr, and Eu could be delayed for 1,000-10,000 years, which is sufficient time for nearly complete decay of those radionuclides. A wetted clay/sand mix also swells appreciably, yielding a nearly impermeable ground water barrier.

Backfill studies in Sweden (340, 346, 376) on bentonite clay, clay/sand mixtures, and barriers of the zeolite mineral clinoptilolite yield similar results: a 0.2-m backfill barrier of clinoptilolite could delay the breakthrough of Cs and Sr in ground water for about 10,000 years: a 1-m thick clay/sand mixture could delay the release of Pu and Np for about 2,000,000 years.

Other ongoing programs (377, 378) are generating additional data for backfill barrier utilization. One activity (377) involves testing the radionuclide sorptive behaviour of various rocks and minerals. Sorption has been shown to be an important potential barrier to nuclide transport (374). The sorptive capacity of several minerals for anionic radioactive chemical species, such as iodine and technetium, is also being studied (373) along with the increased sorption of technetium under reducing conditions is being investigated (365). Others are studying related waste/rock interactions under high temperature, high pressure, aqueous conditions (378) providing sorptive capacity information for various shales and clay minerals.

Currently available data on emplacement hole backfill barrier performance (339, 340, 374, 376), particularly in regard to radionuclide sorptive characteristics, indicate that backfill materials can effectively contribute to the isolation of radioactive wastes in a geologic repository in the presence of brines and other ground waters. Further work on these backfill barriers is in progress (117, 339, 372, 374, 379) for better characterization and engineering development.

II.E.1.6 Waste Package System Component Interactions

Waste package system component interactions (e.g., between the waste canister and the emplacement hole backfill) is of importance for evaluating the compatibility and effectiveness of the entire system. Chemical and/or physical interactions among the waste package/adjacent host rock/ intruding ground water system also must be evaluated in the same light.

Swedish studies (340) on radionuclide release in a repository concluded that a bentonite clay backfill, in conjunction with a thick copper canister (with spent fuel inside) could prevent the release of radionuclides to the host rocks in the presence of granitic ground water, for thousands to hundreds of thousands of years. In those studies the clay barrier served to chemically condition the ground water, reducing its corrosiveness on the canister. The proposed Swedish package was recently reviewed by a National Academy of Sciences subcommittee which judged that its effectiveness to contain the radionuclides in spent fuel for hundreds of thousands of years has been adequately demonstrated (380).

Studies pertinent to mined repositories in salt (338, 372, 374), are in progress to characterize further the interactions between candidate backfill-getter materials and waste container (canister, overpack) alloys. These same studies (338) are investigating dry rock salt/metal interactions and high intensity radiation/salt/brine/metal interactions. Available results indicate that, with the appropriate selection of barrier materials, few or no deleterious effects are observed.

Similar interactions (e.g., waste package system/host rock) are in progress utilizing simulated spent fuel, shales or basalts, appropriate ground water compositions, and hydrothermal conditions (333-335). Potential canister materials will be introduced into these studies this year.

Other accelerated aging (hydrothermal) tests with waste form, host rock, and ground water appropriate to several geologies are under way (316). Further tests are being initiated (313) jointly between Pacific Northwest Laboratory (PNL) and Sandia National Laboratories, utilizing all components of the waste package system and a bedded salt host rock under the same conditions. These tests will be conducted in a hot cell using highly radioactive waste forms. Results from this series of tests will be used for analytical model formulation and verification under expected near-field conditions.

An extensive series of underground experiments in a salt mine, utilizing spent fuel as well as electric heaters, was conducted near Lyons, Kansas, over 10 years ago. This series was designated Project Salt Vault (381). Tests included corrosion studies in salt; radiation effects on rock salt (which showed no radiolytically formed chlorine gas and no detrimental effects on the salt); and measurements of brine moisture migration (which showed the migration rates to be low) (117).

The available in situ data, along with the available results of other studies discussed here, supported by the conclusions reached in studies of individual barrier components, indicate that the waste package system can provide a very effective barrier to the release of waste radionuclides to the near-field host rock, and subsequently to the biosphere.

II.E.1.6.1 Systems Analysis

To evaluate further potential performance of conceptual package designs, some preliminary system analyses have been done (see also II.F.1).

The actual design of a waste package system will depend on the specific characteristics of the repository site for which it is intended, in particular the host rock physical properties and attendant ground water chemistry. However, sufficient evaluations of conceptual package designs have been done, coupled with the knowledge of the basic engineering properties of a wide variety of construction materials, to demonstrate package system feasibility (337, 382). Such work is a continuing activity to refine and narrow the wide range of design options as performance criteria are better defined, materials property data are improved vis-a-vis expected repository conditions, and more information becomes available on specific candidate sites.

Recent design studies (337) have assumed conservative package criteria, e.g., 1,000-year life and a maximum package temperature of 250°C. After examination of a number of options and a wide variety of candidate component materials, four basic design options were developed. The first option consists of two barriers and assumes that the stabilizer is not impervious and does not sorb nuclides (significant flow channels are present). The two barriers are a "corrosion-resistant" metal canister and a tailored backfill. The tailored backfill is a nuclide migration barrier, a cushion to reduce stress durring borehole closure and a water barrier. The canister also serves as a water and nuclide barrier. The second option includes a mild steel canister with the addition of two water/nuclide barriers; an overpack and a sleeve. The concept can be considered a two-barrier design if one assumes that the mild steel may rather quickly corrode. The third option is similar to the second but considers the canister to be corrosion-resistant. The last option considers a corrosion-resistant sleeve with a tailored backfill and either a mild steel can (three barriers total) or corrosion-resistant can (four barriers total).

Even with the rigorous criteria conditions assumed above, it appears that application of any of the four concepts is feasible in either salt or hard rock repositories (337). For salt (brine) application, titanium-, zirconium-, or nickel-based alloys in thicknesses of about 1 in. (2.5 cm) or ceramic or graphite materials, in thicknesses of perhaps 2 to 6 in. (5 to 15 cm) represent conservative designs for long-lived barriers, based on current knowledge of their corrosion and radiation resistance. Cements, to exclude water, and special backfill materials, such as bentonite clay, for water sorption, ion sorption, or ion retardation, may be required in much thicker sections, e.g., from several inches to several feet (tens of centimeters) (376), but have the potential to retard or retain various isotopes for periods from hundreds to many thousands of years.

For application in waters of low ionic strength, pertinent to granite or basalt, for instance, other materials are being considered. Alloys of iron, (e.g., cast iron) in thicknesses of perhaps 2 to 6 in. (5 to 15 cm) and lead- and copper-base alloys in thicknesses of 1 to 4 in. (2.5 to 10 cm) can be added to the list of candidate materials for canisters, overpacks, and sleeves.

Subsequent work used the aforementioned four basic concepts in further systems analyses of package concepts (382). Many combinations of conceptual designs and candidate materials were examined for general repository conditions in basalt, granite, salt, and shale. Although the results of this study must be considered preliminary, the authors concluded that, for all four geologies, packages could be designed with life expectations beyond a thousand years.

II.E.1.6.2 Summary

In summary, spent fuel and its Zircaloy cladding are corrosion-resistant, low-solubility materials. Candidate canister materials are expected to last up to a thousand years, based on current knowledge of their corrosion and radiation resistance. Further, current work on candidate emplacement hole backfill materials has demonstrated the potential to retard or retain certain nuclides for thousands of years. Although not all candidate materials have been evaluated with all potential host rocks and waste packages for specific sites have not been designed, currently available data indicate that appropriate materials and components will be available when required to design and construct the necessary waste package systems.

II.E.1.7 Alternate Waste Forms

As previously described, the waste form upon which this Statement is based is spent fuel (see Part I). Nevertheless, a number of other waste forms have been and are being studied (383, 384). For the sake of completeness, these are reviewed here.

Prior to the decision to defer reprocessing, significant progress had been made in the development and testing of waste forms, such as glass, for wastes generated by commercial reactors. Subsequent to that decision, the emphasis of work on alternate waste forms has shifted to defense-related wastes. The Department is continuing to sponsor work on alternate forms, and it is fully expected that the results and technology developed would be transferable, in large part, to the commercial waste program and existing liquid wastes (379).

Historically, glass, particularly borosilicate glass, has been the major focus of alternate waste form work, and in 1977 it was selected as the reference material (385). Small-scale operating facilities have demonstrated practicality of the vitrification process (379). In addition to U.S. work, studies and pilot plants involving glass are under way in France, Germany, Belgium, and England. Recently, however, more attention has been devoted to other waste forms and studies are being conducted to evaluate their characteristics (384).

II.E.1.7.1 Candidate Waste Forms

The brief descriptions of several alternate waste form candidates given here are followed by descriptions of the current development status of the various forms.

Three of the waste forms being considered for on-site disposal of existing wastes include calcine, obtained when the waste is fired to a mixture of oxides at 300°C-700°C; clay, in which the waste is solidified by mixing with clay to absorb water; and normal concrete, in which the waste is set to a solid in cement. These forms would primarily be candidates for on-site disposal, since they use available technology and are marginal-to-good in leach resistance but offer little intrinsic resistance to transportation accidents.

Four candidate forms for near-term applications include hot-pressed concrete in which interconnected voids and excess water have been eliminated from the normal concrete; pelletized calcine in which the calcine has been agglomerated and its solubility in water has been reduced by firing the waste with various additives; glasses, of which borosilicate glass is the leading contender (but a large number of other glass formulae are possible); and clay ceramics in which the waste-clay mixtures are fired to ceramics. These forms are viewed as current choices for a near-term waste immobilization plant. A considerable amount of evidence indicates that glass is one of the best of these forms.

Three additional forms being investigated include supercalcine in which specific additives are incorporated in the calcine mixture with the objective of producing an assemblage of highly stable, highly leach-resistant minerals (silicates, oxides, molybdates and phosphates) after firing; SYNROC, in which firing or hot pressing is used to produce a similar series of titanate and zirconate minerals; and glass ceramic, in which a waste glass is deliberately partially devitrified under controlled conditions to produce a more stable form.

Finally, three possible composite waste forms are being considered as candidates. They include matrix forms in which pellets of glass, supercalcine, and other waste forms are incorporated in a metal binder; multi-barrier forms in which the individual waste particles are coated with carbon, Al_2O_3, or other impervious materials before being incorporated in a metal or other matrix material; and cermets in which small waste particles are formed in situ in a metal matrix.

These last six forms (and other closely related forms) are the primary candidates for advanced waste form development. Of the six, only glass pellets in a metal matrix are presently well characterized and available for large-scale use. Their development aims at achieving properties consistent with disposal criteria and the possibility of such an achievement is in most cases supported by limited experimental data.

II.E.1.7.2 Alternate Waste Form Development Programs

The Department's wasteform development programs are widely dispersed through the Department's waste processing sites, Department of Energy national laboratories, industrial laboratories, and universities, in order to secure the widest possible input. The following subsection describe the waste forms which are under active development.

II.E.1.7.2.1 Calcine

Calcine waste form development is largely centered at the Idaho Chemical Processing Plant (ICPP) (386), where a long-term program has been pursued to calcine all the plant high-level waste for either interim or permanent storage. Current studies there and at Pacific Northwest Laboratories (PNL) (387, 388) on calcine waste forms concentrate primarily on pelletizing the existing calcines, either for direct disposal or for incorporation in a matrix system (389). Work on calcines also is in progress in the supercalcine program (390) and--calcine as a intermediate--on the borosilicate glass programs at PNL and at Savannah River Laboratory.

II.E.1.7.2.2 Rich Clay

Rich clay and related clay solidification forms are being worked on largely at the Department's Hanford Site as a means for immobilization of Hanford wastes, including in-tank solidification.

II.E.1.7.2.3 Normal Concrete

Normal concrete high-level waste forms have been the subject of a great amount of work (379, 384, 391-394). Major efforts have been at Brookhaven National Laboratory, Oak Ridge National Laboratory, Pennsylvania State University, and Savannah River Laboratory. However, most current work on concrete waste forms is concentrated on in-place applications, such as the Oak Ridge shale hydrofracturing with grout, on the newer higher integrity concretes described below, or on low-level waste applications.

II.E.1.7.2.4 Hot-pressed Concrete

Development of elevated temperature and pressure waste forms is being pursued primarily at Oak Ridge National Laboratory under their FUETAP (Formed Under Elevated Temperature and Pressure) program (370). Pennsylvania State University has developed hot-pressed concrete forms (371, 395, 396) and other tailored cement formulations (397).

II.E.1.7.2.5 Borosilicate Glass

Borosilicate glass is generally accepted as the best currently available immobilization form for high-level nuclear waste. It is the most well-developed form and is continuing to receive the largest share of the development effort. The U.S. effort is primarily focused at Pacific Northwest Laboratory and at Savannah River Laboratory (360, 398-401). Work on adapting the borosilicate glass to their particular waste compositions is also under way at each of the waste processing sites (402, 403).

II.E.1.7.2.6 High-Silica Natural Glasses

The high silica natural glasses known as obsidians and tektites have persisted for long periods in both terrestrial and lunar environments. However, these glasses melt at about 1,600°C, a temperature high enough to drive off most of the ruthenium and cesium radionuclides from the waste during processing. Several proprietary processes are being investigated for low-temperature formation of high-silica glasses containing high-level wastes (404, 405).

II.E.1.7.2.7 Clay Ceramics

For clay ceramics, adding aluminum silicate clays such as kaolin or bentonite to the waste typically produces an insoluble cancrinite type material. This material can be fired to a nepheline-like ceramic. Some consideration is being given to those materials; however, most of the attention is focused on the more-advanced ceramic analogs of long-lasting natural minerals in the tailored ceramic (including supercalcine) and SYNROC programs considered below.

II.E.1.7.2.8 Supercalcine

The tailored ceramics are under study at Pennsylvania State University (406-409). Working in cooperation with the Pacific Northwest Laboratory, investigators added various chemical modifiers to liquid or solid waste prior to calcining, sintering, or hot pressing to produce synthetic analogues of stable natural minerals; calcining is followed by hot pressing or sintering to ceramic waste forms. They are continuing this work in cooperation with PNL and the Rockwell International Company (410).

II.E.1.7.2.9 SYNROC

These materials, assemblages of synthetic titanate and zirconate minerals proposed for use as waste forms, are based on natural analogues that have persisted in nature for very long times and that can be sintered or hot pressed to ceramic forms (411). Lawrence Livermore Laboratory is working on SYNROC development. In addition, SYNROC-type compositions are being studied in a number of the other U.S. waste form programs on an exploratory basis.

II.E.1.7.2.10 Titanates, Niobates, and Zirconates

Certain of these compounds were developed by Sandia National Laboratories (412) as mineral ion exchangers. A small program is in progress to determine the practicality of hot pressing or sintering them to waste forms. These materials are also being considered as engineered barriers around the waste forms.

II.E.1.7.2.11 Glass Ceramics

One form of glass ceramic can be made by sintering or hot pressing the mixture of waste and glass frit (413) rather than by melting as in normal glassmaking practice. The resulting lower temperature processes have some attraction in reducing radionuclide volatilization and chemical corrosion; they have received limited attention for the fluoride-containing wastes at ICPP (403, 414, 415). The more common forms of glass ceramics are formed by controlled devitrification. PNL is pursuing a small program in this area in cooperation with a larger program in the Federal Republic of Germany.

II.E.1.7.2.12 Matrix Waste Forms

Metal matrices can be used with most of the waste forms discussed above (such as calcines, concretes, glasses, ceramics, and artificial minerals); the waste forms can be of small size and dispersed in a metal matrix for better heat transfer, reduced frangibility, easier process sampling, and additional leaching barriers. Low-melting alloys of Pb or Al can be cast around the waste particles, while high-melting metals can be sintered around the particles at temperatures of about two-thirds of their melting temperature. Metal-matrix waste form work in the United States is primarily concentrated at Argonne National Laboratory and Pacific Northwest Laboratory (416, 417).

Matrix waste forms with additional barriers can be made by coating the waste particles with impervious materials such as carbon, alumina, or silicon carbide before placing them in the matrix. Such coatings provide additional barriers against waste leaching and also allow the use of high-temperature matrix-forming processes by reducing radionuclide volatilization. Concretes, sintered ceramics, and other materials can be used rather than metal as the matrix when coated particles are used. The primary U.S. effort on these multibarrier forms has been performed by Battelle Memorial Institute in their Pacific Northwest and Columbus Laboratories (390, 418). Consideration is also being given to applying the coated-particle technology, developed by General Atomic Company for their high-temperature, gas-cooled reactors, to the multibarrier forms.

II.E.1.7.2.13 Cermet

These cermet high-level waste forms are a particular matrix form in which very fine waste particles are dispersed in a metal matrix, usually by in situ precipitation. Oak Ridge National Laboratory (419, 420) is developing a particular waste cermet in which the wastes (and additional metal formers) are dissolved in urea, and the metal formers are reduced from the solution to form a Hastelloy-like (trademark of Cabot Corporation) alloy containing finely dispersed nonmetal waste particles.

II.E.1.7.3 Alternate Waste Form Summary

In summary, work on the development of alternative waste forms is actively under way, with emphasis on defense wastes. While the representative waste form discussed in this statement is spent fuel (see I.A), the evolving alternate waste form technology could be applicable to other forms of commercial wastes, should the need arise. Also, supported by the current (mainly defense-related) work, this technology provides the Department's waste management program a high level of flexibility and a large number of options.

II.E.1.8 Waste Package Summary

This discussion of the waste package, which is one component of the man-made systems of mined geologic disposal, has described possible package components and candidate materials from the standpoint of the Department's current state of knowledge and the programs in place or planned for their further development. From the discussion, it is obvious that much remains to be learned about individual package components and their interactions within the waste repository environment. Nevertheless, a large body of information is available and it continually is growing. The large number of options open to the NWTS Program due to the diversity of the studies described provides a large measure of confidence that several acceptable waste package combinations will be identified. Based on currently available knowledge, it is expected that the waste package system will meet the stated criteria. The waste package will be a multibarrier system in itself to enhance confidence that the total package system will function as an effective barrier to radionuclide release to the near-field host rock. The waste package will work in concert with other man-made system components and with the natural system to provide additional multibarrier redundancy and attendant assurance of adequate containment and isolation.

II.E.2 Repository System

As discussed in Chapter II.C, the repository system will provide for the receipt, inspection, transfer to the underground, emplacement, and containment after closure of radioactive waste. The waste will be emplaced in a manner that would allow retrieval, if necessary, during the operational phase of the repository. Provisions for decommissioning and monitoring will also be made.

In the design of a repository, consideration will be given to both long-term containment and isolation and operational factors.

As stated in Chapter II.C, the surface facilities of a repository are similar to those now used in the nuclear industry. Radiation protection practices in the repository, therefore, will be similar to those used in other nuclear facilities and thus are not discussed here. Radiation protection, as a factor in the overall assessment of repository performance, is discussed in Chapter II.F. Repository support facilities and underground workings are also similar in many ways to those common to the mining industry. Therefore, issues not uniquely related to radioactive waste repositories, such as the construction of support facilities, are not discussed here.

The long-term isolation requirements important to repository structure are:

1. Limiting the impacts of the development and operation of the repository on the containment and isolation capabilities of the natural and manmade systems.

2. Enhancing the natural containment and isolation capabilities of the repository system through the use of engineered barriers.

The potential long-term impacts would arise from the following factors:

1. The excavation of underground disposal areas.

2. The introduction of heat generated by the waste.

3. The introduction of radiation generated by the waste.

4. The introduction of penetrations such as exploration boreholes, shafts, and tunnels into the rock mass.

The possibilities for enhancing the containment and isolation capabilities of the natural system include, principally, the use of sealing and backfill materials that will retard radionuclides.

During the operational period, the stability of the structure must be ensured throughout the waste emplacement and retrievability phases. Stability is discussed in subsections II.E.2.1 and II.E.2.2.

Each of these factors is discussed in terms of (i) its impacts on the natural system and importance to repository design, (ii) the requirements placed on repository design to avoid or mitigate impacts on the system or to take advantage of measures that would enhance isolation, and (iii) the status of the Department's ability to meet these requirements.

II.E.2.1 Excavation and Underground Development

This subsection discusses two potential impacts that must be considered in the excavation of a repository: (i) fracturing around the perimeter of the tunnels and rooms and (ii) impact on in situ stress states and its implications for long-term containment and isolation.

II.E.2.1.1 Potential Impacts

II.E.2.1.1.1 Fracturing Due to Excavation

Conventional mining techniques, such as drilling and blasting or continuous mining, will be used in developing the underground facilities of a repository. Continuous mining can generally be used for softer rocks such as salt (421). The use of drilling and blasting is necessary in the harder rocks (422, 423). When the explosive charges used in the drilling and blasting techniques are detonated, the local rock is shattered and the broken material is collected and transferred to the surface or used as backfill underground. Continuous mining systems employ a machine to continuously remove the material from the development area. As the material is removed, it is either continuously transported away from the operating face to the surface or used as backfill material (423, 424).

During the excavation process, the balance of the forces present in the geologic formation is altered. As the formation reestablishes equilibrium, effects such as localized cracking can develop. Therefore, fracturing around the perimeter of the excavated areas can result from the direct effects of blasting or from stress concentration effects around the excavation and reestablishment of equilibrium (423). Fracturing around the excavation, if extensive, may provide a potential pathway for ground water. The extent of this fracturing depends on the rock type, the extent of natural fracturing, the depth, and the excavation technique (423).

II.E.2.1.1.2 Mine Stability

The excavation of rooms and tunnels underground will induce a new stress state and displacement field in the host environment. The nature of these stresses and displacement fields depends on the cross-sectional geometry of the excavation, the layout of the tunnels and rooms, and the extraction ratio (the ratio of the volume removed to the volume remaining) (423).

Consideration of these factors in the design of the excavations is of importance to safety during the operational phase. While the objective of conventional mining is to achieve the maximum amount of extraction (removal of ore, mineral, etc.) at the lowest cost, the design of the underground waste repository is based on extracting only what is necessary to provide sufficient space for waste emplacement. Extraction ratios are therefore chosen in the 10%-20% range, whereas in a conventional mining operation the extraction ratio can be as high as 90% (425). With these low extraction ratios, mine safety during operation is enhanced. (See Chapter II.E for further discussions of operational safety.)

Of interest to long-term containment and isolation is the possibility of subsidence in the strata overlying the repository, which might lead to adverse perturbations in the hydrologic regime (426). Mine stability is also discussed in Subsection II.E.2.2.

II.E.2.1.2 Excavation Requirements

Excavation requirements for the repository structure are:

1. Excavation of the repository must not introduce potential ground water flow paths by virtue of fractures caused during excavation. Therefore, if blasting techniques are used, control must be exercised in designing blasting patterns and selecting charge sizes and sequences of detonation. The low extraction ratios and room geometries anticipated in repositories should limit the extent of fracturing due to stress concentrations (423).

2. Extraction ratios will be maintained at conservatively low levels to ensure that repository stability satisfies the appropriate thermal and thermomechanical limits (426).

II.E.2.1.3 Status of Knowledge Relative to Excavation

Vast experience has been gained in the excavation of various kinds of underground facilities. Fracturing during drilling and blasting operations is limited by controlling such parameters as the size and type of charge, the configuration of drill holes, and the sequence of detonation. Controls of these types are used extensively in the excavation of underground facilities intended for storage purposes and for long-term operations (427); examples are caverns for compressed air and natural gas storage. In situ tests are in progress to confirm their suitability for the excavation of mined geologic repositories (428). It is believed that no further technological advances are needed in this area (429).

The extent of subsidence of the ground surfaces will be reduced by employing relatively low extraction ratios and backfilling the excavated areas after waste emplacement. Although it is impossible to backfill an area completely to the original density of the rock mass, the backfill material does contribute to the support of the overburden and limits the subsequent amount of subsidence. The backfilling of excavated areas in mines is a standard practice for achieving long-term stability. This backfilling, in conjunction with the already low extraction ratios, makes the volume of the excavations that will remain unfilled so small in relation to the remaining host material that subsidence will not be a significant problem (430).

II.E.2.2 Thermal Effects

Limiting the impacts of heat generated by the waste is a principal consideration in the design of a repository (431, 432). The repository must provide assurance that thermomechanical and thermochemical interactions will not endanger the structural stability of the repository or the integrity of the host medium, cause significant impacts on the hydrologic properties, or lead to premature degradation of the waste package.

II.E.2.2.1 Potential Thermal Impacts

The introduction of heat and the associated temperature rise in the repository could affect the stability of the structure (thermomechanical impacts), which is only of concern in the long term if it adversely influences the structural integrity of the waste package or the hydrologic properties of the surrounding rock mass. Heat will also affect the chemical interactions between the waste package and the fluids contained in the rock in the near field (thermochemical impacts). Finally, there may also be thermal impacts on the hydrologic system (thermohydrologic impacts). These impacts are discussed in the paragraphs that follow.

II.E.2.2.1.1 Thermomechanical Impacts

The introduction of heat into the natural system will increa the temperature and thus induce stresses in the host rock and surrounding media (433, 434). These stresses will be superimposed on the existing stresses and must be considered in design to ensure structural stability of the repository. The heat generated by the emplaced waste will cause the rock mass to expand, thus inducing surface uplift. In the long term, as the heat is dissipated, the surface will subside. Displacement of the overlying rock mass may cause fracturing in the rock, thereby giving rise to perturbations in the hydrologic flow regime. In addition, heat may modify the thermal and mechanical properties of the rock; for example, an increase in temperature will enhance the ductility of a rock but reduce its ultimate strength.

II.E.2.2.1.2 Thermochemical Impacts

The thermochemical impacts of principal interest in repository design are those that would accelerate the degradation of the waste package and the migration of radionuclides away from the package. The introduction of heat into the system will change the environment in which the waste was placed. The design of a waste package capable of withstanding the heat-altered environment is discussed in Section II.E.1.

One of the possible thermochemical phenomena is the influence on the waste package of incursion of fluids into the very near field through, for example, brine migration up the thermal gradient in salt or the decomposition of hydrated minerals and the liberation of water, as might occur in some argillaceous rocks. Incursion of water is less likely in granites and basalts, owing to the limited quantities of contained fluids and hydrated minerals (433, 435). However, waste package interactions with water in fractures in these rocks will be influenced by the temperatures in the repository. For example, metal corrosion is accelerated in the presence of heated fluids.

II.E.2.2.1.3 Thermohydrologic Impacts

In addition to the thermomechanical impacts on the hydrologic system and the thermochemical introduction of free fluids into the very near field, the heat generated by the waste may also perturb hydrologic flow regimes by affecting fluid pressures, decreasing fluid viscosity, or inducing convection cells (435, 436).

II.E.2.2.2 Thermal Requirements

The magnitudes of the potential impacts described above are dependent on the temperatures reached in the various components of the repository system. By restricting temperature, unacceptable impacts can be avoided. Therefore, the following requirements will be specified:

1. The temperatures in the mined geologic disposal system will be restricted to limits within which thermal impacts on the system can be (i) predicted and (ii) shown to cause no significant degradation in the system's containment or isolation capabilities (437).

2. To ensure that the potential impacts of the phenomena described in Subsection II.E.2.2.1 are controlled, limits will be set for the maximum temperatures of the (i) waste package, (ii) sealing materials, (iii) rock mass, and (iv) neighboring aquifers (438-441).

3. Temperatures in the repository during the operational period will be limited to maintain stability, thereby protecting the repository personnel and retaining the option of waste retrieval (439-441).

Temperature limits are necessary to maintain structural and geochemical integrities of the waste package and sealing materials; to ensure protection of the structural integrity of the rock mass and to minimize changes in permeability; and to prevent irreversible perturbations of the ground water flow systems (438).

Temperature limits are highly dependent on the characteristics of the waste package, the site, and the host rock. Temperature limits can be determined for different generic host formations but will be reevaluated and refined on the basis of site-specific data for proposed repository sites (438).

II.E.2.2.3 Status of Knowledge About Thermal Effects

The status of knowledge about thermal effects is presented in this section by describing the design process for minimizing the impacts of thermal effects and presenting two examples. The first example is typical of the development of thermomechanical design specifications; the second example is a study of fluid migration, which is one of the thermochemical phenomena.

II.E.2.2.3.1 Design Process

In the process of designing a repository to minimize the influence of thermal impacts, three steps are followed (438, 441-443):

1. The thermal limits are prescribed in terms of allowable temperatures, stresses, and deformations in the system. These limits are based on an understanding of the physical processes controlling the behavior of the system and on the observation of the response of the components of the system in laboratory and in situ tests.

2. Models that analyze the temperature, stress, and hydrologic perturbations in the rock mass due to facility excavation, spent-fuel emplacement, and sealing are used to determine the design specifications (e.g., thermal power density) that must be applied to ensure that the thermal limits are not exceeded.

3. A design is developed that satisfies the prescribed thermal limits and other applicable design constraints.

The status of the capability to proceed through each of these steps is described in the paragraphs that follow.

Qualitatively, temperature limits are based on a physical understanding of the processes and characteristics of potential deformation and failure in the waste package, sealing materials, and the rock mass over the range of environmental conditions expected in the repository system. Quantitatively, limits are established from the results of laboratory and in situ tests and the observations of the behavior of natural geologic systems under conditions of elevated temperature and pressure. Furthermore, the limits are quantified for each geometric zone of the repository and for time periods of repository performance.

The perturbations of temperature and stress that are induced in a rock mass by repository excavation, spent-fuel emplacement, and sealing, are calculated by means of mathematical models. They are dependent on both time and space. These models, described in Section II.F.1, require the following data for application to a particular situation:

1. The heat-generating characteristics of the spent fuel (see II.C).

2. The geologic, geotechnical, geochemical, hydrologic, thermal, and mechanical characteristics of the host-rock mass (see II.D).

In the design of the repository, the maximum temperatures in the waste package, the engineered barriers, and the rock mass will be controlled by limiting the thermal loading in the repository. This can be accomplished by (i) aging the waste to reduce the amount of heat that is generated after emplacement, (ii) limiting the number of spent-fuel assemblies per canister to reduce the thermal output, or (iii) limiting the number of waste canisters per acre of the repository. Each of these options will be considered on a site-specific basis. The lower portion of Figure II-22 (444) shows the reduction in thermal power per assembly of BWR or PWR fuel with fuel age.

The thermal output of the fuel can be reduced more than 50% by aging for a period of 10 years after removal from the reactor. The upper portion of the figure illustrates the number of fuel assemblies of a certain thermal power that can be emplaced per acre to achieve the desired thermal power density.

II.E.2.2.3.2 Example: Development of Thermomechanical Design Specification

In early design studies for HLW repositories in bedded salt, temperature limits were quantified for the very near field and the far field. In the very near field, the maximum temperatures that could be sustained in certain volumes or regions of salt around the canisters were quantified on the basis of observed salt deformations at elevated temperatures in laboratory tests. Far-field temperature limits were established on the bases of allowable temperature rises in aquifers and on the boundary of the repository buffer zone. With thermal models, a thermal power density of 150 kW/acre (37 W/m^2) for 10-year-old high-level waste was found to satisfy the temperature limits for the assumed bedded-salt formation (445).

In more recent design studies for spent-fuel repositories in generic bedded and domed-salt formations, limits were defined for the thermomechanical behavior of the various rock formations on the basis of a well-established strength failure criterion for rock. In effect, the limit specified that none of the rock formations could experience strength failure as a consequence of stresses induced by heat production from spent fuel disposal. Conservative values of the strength parameters were selected from data for generic rock types. Thermal and thermomechanical calculations indicated that at a thermal power denisty of 75 kW/acre (19 W/m^2), the failure criterion is satisfied, except for a localized region of rock salt in the far field. Therefore, for the limit assumed in this study, thermal power densities less than 75 kW/acre appear to be acceptable for repositories in domed and bedded salt (446). Subsequently, these calculations were repeated for the purpose of assessing the impact of variations in the thermal and mechanical properties of the rock formations, including the presence of joints in the nonsalt formations, and in the age of the spent fuel (28).

Design studies like those just discussed have also been performed for spent fuel repositories in granite (441, 443) and basalt (448, 449). Temperature and thermomechanical limits have been prescribed for the waste package, the engineered barriers, and the rock mass for the three repository geometric zones of interest in relation to the various time periods of repository performance. In order to meet these limits, computer models and generic characterization of the rock masses have been used to determine the required thermal power density, canister spacing, extraction ratio, excavation geometry, and artificial support requirements. The values thus determined provide design specifications, developed from conservative premises so as to ensure safety in the performance of the repository over the long term. However, these specifications must be coordinated with operational considerations during the excavation, spent-fuel emplacement, and closure periods, and with requirements for retrievability in the event that option is exercised.

The above discussions demonstrate that a rational and consistent approach to repository design is being followed in a rigorous and defensible manner. The temperature and thermomechanical limits are being refined by means of interrelated programs of laboratory and in situ tests and by analyses of the long-term performance of natural geologic systems. Programs of model development, refinement, and verification are proceeding in concert with the above efforts (450, 451). Specific examples of the results of these studies are cited in the following discussions.

Accurate spatial and temporal calculations of temperatures can be provided upon specification of temperature-dependent thermal properties that have been measured for rocks being considered as repository host media (114, 452-454). Field measurements have been performed for locations where in situ tests are in progress or planned (452, 455); they have also been performed to support modeling and design efforts for regions under investigation as potential repository locations (442, 448, 456). These model calculations have been or will be compared with the results of field and laboratory tests (see II.F.1). These techniques will make it possible to specify temperature limits as well as thermal power densities and canister power levels that will ensure that temperature limits are not exceeded in the repository system. Examples of the expected temperatures in basalt, salt, and granite are given in Chapter II.C.

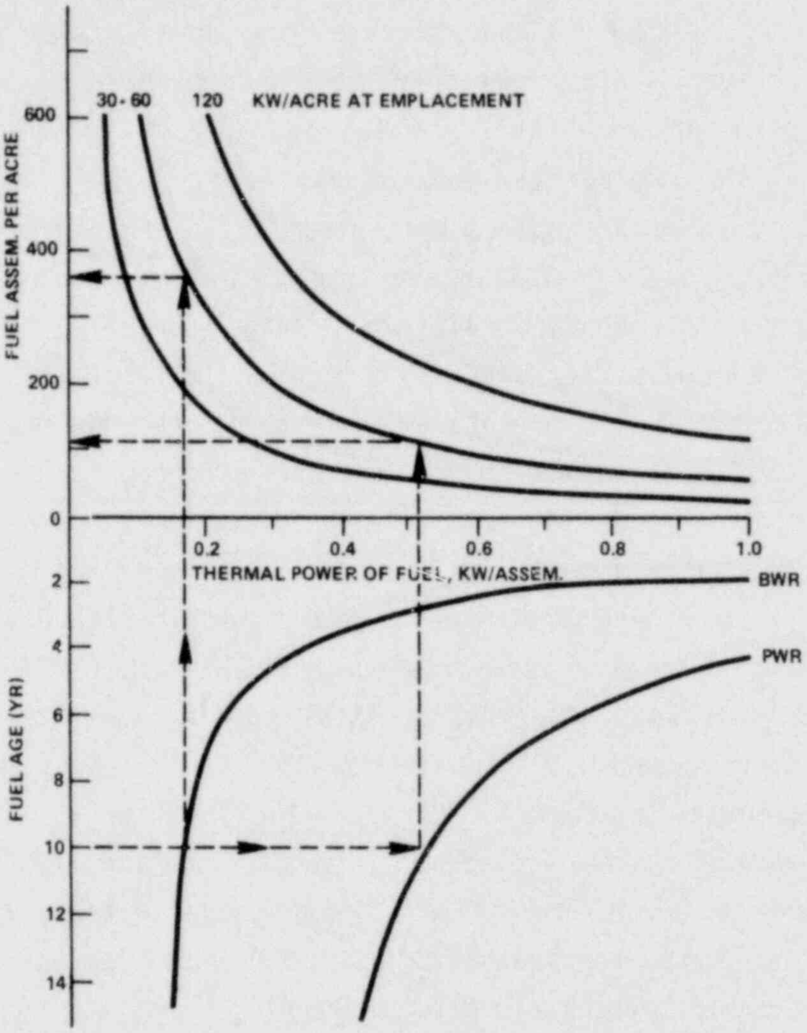


Figure II-22. Thermal Power From a Spent Fuel Assembly As a Function of Time

Source: (Reference 444) Bechtel National, Inc., National Waste Terminal Storage Conceptual Reference Repository Description, Vol. III, ONWI/SUB/79/E512-01600.16, Battelle Memorial Institute, Columbus, OH, September 1979

Near-field thermomechanical calculations have addressed primarily the conditions that affect retrievability. These include room temperatures, room closure, and thermal stresses in the rock surrounding the room. Numerous calculations of temperatures in pillars have been performed (457, 458). Room deformations in salt, which incorporate transient and secondary creep behavior, have also been extensively analyzed (451). Calculations for periods of 25 years after emplacement, assuming no backfilling of the emplacement rooms and no use of additional structural supports, have been performed for basalt, granite, and salt (448, 459, 460). Stability criteria for hard rocks involve the use of strength failure criteria, including the influence of joints, at the computed temperature. For salt, the thermomechanical limits are based on the room closure so as to allow retrievability without a need for remining.

Calculations of surface uplift and perturbations in time regional temperature field, including expected surface temperatures, have been performed for generic repository environments in basalt (448), granite (443, 461, 462), and salt (432, 446, 447). These efforts have considered such factors as the thermomechanical behavior of the rock masses, parametric variations of the in situ stresses, rock properties, waste age and thermal power density, sequential waste emplacement, shaft location with respect to the waste emplacements, and perturbations of the ground water system. Parametric studies of these types form rational bases for establishing repository design specifications for site-specific situations.

II.E.2.2.3.3 Examples of Investigation of Thermochemical Phenomena--Fluid Migration

Fluids are a common component in the formation of rocks, whether as a chemical constituent or as a vehicle for deposition. Fluids contained in rock can occur in three ways: as intracrystalline accumulations, (e.g., fluid inclusions or negative crystals in evaporites) as intercrystalline accumulations along crystal and grain boundaries, and as waters of crystallization, (i.e., water chemically bound to minerals in the rock).

Contained fluids will be of concern if they are mobile, if they can interact with the waste package, or if they can transport radionuclides from the package into a hydrologic system. On being heated after waste emplacement, any of these three forms of contained fluids may be mobilized. Fluid mobilization depends on the rock type and the temperature and temperature gradients to which it is exposed. The mobilization of intracrystalline and intercrystalline fluids in a thermal gradient has been studied for certain evaporites (463, 464). A large body of information is available on the thermodynamics of phase alterations and the subsequent release of waters of crystallization. There has also been considerable study of changes in the physical state of rocks in which the release has occurred (465). The importance of each type of fluid varies widely among the various rock types under consideration for repository applications, as discussed in the following paragraphs.

Granite and Basalt. The only significant fluids that can interact with the waste package in either basalt or granite will be those transmitted through fractures with finite hydraulic conductivities, or those that might be contained in any clays within the fractures. Apart from the fluids contained in the fractures, neither intracrystalline fluids nor waters of crystallization will be mobilized at the temperatures expected in repositories (less than 300°C) (465).

Bedded Salt. Bedded salt can contain from less than 0.1% up to approximately 1% by weight of fluids (116) in all of the forms discussed above: fluid inclusions, intercrystalline fluids, and water of crystallization (116, 117). Studies of fluid migration have been performed during Project Salt Vault and for salt from the Los Medanos site in southeastern New Mexico for the Waste Isolation Pilot Plant (WIPP). A general plan for evaluating fluid migration has been formulated for the WIPP project (466). Factors affecting brine migration have been described (467). Much of the information available on studies through 1978 on brine migration in salt has been summarized to support further experiments (468).

The information on the migration of fluids has been evaluated and used to correlate the velocity of migration with temperatures and temperature gradients (469). These correlations have been used in the MIGRAIN code (see II.C). Most of the information analyzed was based on observations of migration within single crystals. Therefore it was proposed that the greatest uncertainty in the correlations was in the ability of inclusions to migrate across crystal boundaries. Further experimental evidence indicates, however, that much of the intracrystalline fluid is trapped in the crystal or at the crystal boundary, which limits the quantity of fluid available for release.

During Project Salt Vault, it was observed that the release of fluid into heater holes increased markedly after the electrical heaters used in the experiments were shut off and the salt cooled (116). It was suggested that this phenomenon was attributable to an alteration in the state of stress in the heated salt, which allowed fluids trapped in the salt to escape (469) as the salt contracted.

The increase in fluid influx upon heater shutdown has been observed in subsequent experiments (118, 119). In the Salt Block II experiment (118), a 1-m^3 salt block, taken from a potash mine in southeastern New Mexico was heated by a 1.5-kW heater. A total of 110 g of fluid was collected in about 100 days of testing. Temperatures of 200^{o}C and gradients of 10^{o}C/m were observed. More than 40% of the fluids collected was released during the shutdown phase. Mineralogical analyses of the salt block are in progress (470). In one examination, optical microscopy revealed the presence of elongated tracks of fluid inclusions, oriented radially with respect to the heater hole. Many of these tracks are terminated at crystal boundaries (at the "heated" end) or within the crystal in which the inclusion originated. Carnallite encrustations were found on the surface of the hole containing the heater.

The results of experiments completed to date can be summarized as follows:

1. The presence in the salt-block sample of tracks that terminate within the crystal or at crystal boundaries supports the theory that fluid inclusions may not be able to cross crystal boundaries. Experimental evidence also indicated that liquid migration might occur along grain boundaries.

2. The formation of carnallite on the heater hole surface of the salt block indicates that some brine (which contains many of the mineral constituents of carnallite) did in fact reach the heater in the liquid phase.

3. The increased influx of fluid upon heater shutdown in all the experiments suggests that a significant portion of fluid collected may be attributable to experimental conditions, such as the relatively rapid cooling of the salt, which are not expected in a repository environment.

4. An additional mechanism of fluid migration has been proposed: vapor phase transport. It has been suggested that this effect is also a cause of increased fluid influx due to experimental conditions (464).

Further tests are planned to resolve whether these mechanisms (brine migration or vapor phase transport, or both) will apply under actual repository conditions.

Dome Salt. In general, dome salt contains fewer fluid inclusions than bedded salt and less total fluid (approximately 0.2%) (471).

Shales. The amount and type of fluids contained in shales depend on the mineral composition of the clay and the diagenetic and metamorphic history. Heating and subsequent dewatering in shales can produce fractures (465, 472, 473). Fluids can also be released from shales because of mineralogical changes induced by heating.

Tuffs. These rocks can contain fluids not only in isolated pores but also in minerals, such as secondary zeolites and clay alteration products. The release of fluid can occur at temperatures on the order of $100^{\circ}C$ to $200^{\circ}C$ (474). Further information is required to assess the impact fluid migration or release in tuffs.

Summary. The migration of fluids around waste canisters has been suggested as a mechanism which might lead to waste-package degradation.

In salt, the observed and calculated quantities (II.C) of fluid reaching an emplacement hole is small compared to that necessary for significant degradation of the waste package. In addition, numerous engineered features are available to limit fluid interactions with the waste form (II.E.1). While fluid may be present in granites and basalts because of flow through fractures, there is relatively less concern about mobilization of contained water. Experimental observation of bedded and domal salt indicate that small quantities of fluids are released. It has been suggested that the release may be due primarily to experimental conditions (e.g., dry boundary conditions and changing stress states) that do not represent the repository conditions around a spent fuel canister. Further experimentation will be conducted to determine which mechanisms actually apply in the repository. Shales and tuffs can contain large quantities of water but require additional investigation before any conclusion about the possible impact of such can be stated.

II.E.2.3 Radiation Effects

The effects exerted on the host rock by irradiation have generally been considered to be of secondary importance. The impacts that do result occur within the first meter of host rock surrounding a canister (475) and thus are primarily of concern to waste package design rather than the repository (476). The status of knowledge on radiation impacts on the host rock and on its interactions with the waste package is described below.

II.E.2.3.1 Status of Knowledge on Radiation Effects

The radiation levels for a reference spent-fuel assembly were given in Chapter II.C. The maximum gamma-ray surface dose rate for 10-year old fuel given there is 23,400 rad/hr. The energy deposition in surrounding rock can be calculated with radiation transport models. The initial deposition rate in salt has been estimated to be 7 x 10^6 rad/yr from gamma rays and about 8 rads/yr from neutrons (432). The attenuation of the gamma rays in the salt is such that a tenfold reduction in dose occurs for each 15 cm from the exposed surface. Hence the salt 30 cm (1 foot) from the canister would experience a dose rate approximately 100 times lower than that at the surface. After 10,000 years, the accumulated dose in the salt nearest the waste canister is on the order of 10^9 rad (432). Similar doses would be observed in other rocks (e.g., granite and basalt) because the gamma-ray absorption characteristics do not vary greatly for the principal constituents of the various rock types.

The possible effects of neutron and gamma radiation in the repository and its interaction with the waste would consist of exposure to operating personnel, radiation chemistry effects such as radiolysis, and radiation damage in the local host rock. The potential for criticality in a spent-fuel repository is also a consideration.

The radiolysis of fluids that could contact the waste canister is one of the principal factors that could affect canister longevity (477). It has been demonstrated that the radiolytic production of various ions, which increase the oxidation potential of solutions, enhances corrosion rates (478). In addition, the radiolytic production of gaseous phases may alter fluid migration rates (479).

Radiation damage in the host rock can produce stored energy in the form of defects introduced into the crystal lattice (480). The effects of stored energy in a salt repository have been analyzed (475), and no serious consequences have been identified. The amount of stored energy accumulated for a given dose, its release under thermal annealing, chemical reactions upon aqueous dissolution of the salt, and the alteration of mechanical properties have been studied (479, 481, 482). This work has confirmed that appreciable amounts of gamma-radiation energy can be stored under certain exposure conditions and that thermally activated annealing takes place at elevated temperatures. The rate of annealing is such that negligible amounts of energy will be stored in salt in a repository where the salt is at temperatures above about 150°C. Thermally activated annealing in rock salt at temperatures below about 150°C was not found (481). The results of the measurements of energy stored at irradiation temperatures between 30°C and 150°C, together with theoretical calculations, showed that the maximum stored energy that would be formed in salt with no annealing whatsoever would be about 1 cal/g at the point of maximum dose. No means by which the stored energy could be released abruptly has been identified, but even if it did occur, the release of stored energy is not expected to pose a significant hazard (432).

An additional aspect of the stored energy is the generation of hydrogen, which would take place if radiation-damaged salt is dissolved in water; at standard temperature and pressure, about 0.1 cm^3 of hydrogen gas is generated per calorie of stored energy. The impact of hydrogen gas generation can be counteracted by the design and operation of a repository during the operational phase; the hydrogen will diffuse away during the long term.

To date, most of the laboratory and theoretical studies have concentrated on the effects of radiation on salt. The information available on radiation effects on salt and on other geologic formations of interest for waste disposal has been compiled (483). It is desirable, at this point, to conduct in situ tests to determine the effects of radiation on interactions between the host rock and the waste package and to ascertain whether deleterious reactions occur due to synergism between the heat, radiation, and chemical interaction with the package (484).

The potential for developing a critical configuration with spent fuel in the repository has been evaluated for both the operational period and the period after decommissioning.

Neutron multiplication factors (k_{eff}) have been calculated for several repository storage configurations (485-487), including clustered canisters and various canister spacings. The most reactive credible configurations were assumed: no initial burnup and full reflection by water or brine. The neutron multiplication factor calculated for a two-row, 4 by 5.5-ft configuration of 50 canisters surrounded by brine is less than 0.6. A neutron multiplication factor of 1.0 is necessary to achieve criticality. It is concluded from these studies that no configurations approaching critical conditions can be expected.

The potential for criticality if waste dissolution were to take place has been considered for salt. A many fold reconcentration of fissile material would have to occur in the repository before a critical mass could form. Such reconcentration would require extensive dissolution of the salt and the waste. After dissolution, other unlikely processes would have to act on the waste, selectively removing fissile nuclides from their surroundings and collecting them into a separate mass. The only known assembly of fissile material into a critical mass by geologic processes occurred in the Oklo uranium deposit in Africa (488, 489). The processes that assembled the critical mass operated over millions of years on a body of underground uranium ore that was much richer in fissile material (approximately 3%) than the spent-fuel in a repository (approximately 1%).

Further studies will continue to investigate hypothetical scenarios (490, 491) describing the reconcentration of fissile material. If any of these scenarios appears to have an appreciable probability of occurring, additional calculations will study their effects. The formation of a critical mass would not necessarily have serious impacts on a repository (490). Calculations investigating criticality and its consequences will continue. In view, however, of the self-limiting behavior of a critical assembly and the reconcentration required to produce it, nuclear criticality is not expected in a spent-fuel repository.

II.E.2.4 Repository Penetrations

In the development of a repository, penetrations will be introduced into the natural system in the form of exploratory boreholes and repository shafts. In addition, a proposed repository site may have some boreholes or wells from previous exploration or resource-recovery activities. Such penetrations are potential paths for ground water flow through the repository. Ground water flow may occur along rock/seal interfaces or through fractured rock adjacent to seals (492).

II.E.2.4.1 Potential Impacts of Repository Penetrations

A major consideration in repository sealing is the extent to which the site must be brought back to its original undisturbed state (492). This question encompasses such concerns as the allowable permeability of the seals as a function of time, the placement of new boreholes, the techniques used in shaft construction, and the location and characterization of existing penetrations. These concerns are addressed in the next section.

The characteristics of the environment in which plugs and seals are to be placed influence their design, material composition, placement, and performance. The important characteristics are (i) stratigraphy and hydrogeology; (ii) load conditions, including temperature and pressure; (iii) rock properties; (iv) borehole and shaft characteristics (e.g., size and lining); and (v) the geochemical environment. Although some broad generic characterizations can be made, most of these factors must be determined individually for each site.

Stratigraphy and hydrogeology at a particular site are important in determining the sequence of materials used in borehole or shaft plugs or seals. Temperatures and pressures, which themselves depend on host-rock behavior, affect the behavior of both seal materials and the surrounding rock, the design and placement of seals, and the rate of geochemical reactions and, thus, the longevity and durability of seals. The temperatures and pressures of concern include both those under the natural conditions at the repository and those induced by repository construction and waste storage. General values for natural conditions of temperature and pressure can be determined from available data, site characterization studies, and analytical models. Changes in temperature and pressure conditions induced by construction and storage are influenced by the actual repository design. The seal design will also take into account the properties (thermal, hydrogeologic, mechanical, chemical) of the surrounding host rock.

Finally, the design and placement of seals are influenced by the characteristics of the penetration, including size and age and the presence of casings, linings, and previous plugs or other material. These factors will be determined individually for each site and will be considered in design (493).

II.E.2.4.2 Penetration Sealing Requirements

Penetrations into the host-rock system will be sealed to prevent significant amounts of ground water from entering the emplacement region and to prevent radionuclides from reaching the biosphere in quantities that would exceed acceptable levels. Penetrations must therefore be sealed to exclude water, retard radionuclide migration, and prevent communication between aquifers to the extent necessary for adequate isolation (494).

Penetration sealing requirements for the repository are:

1. Adequate Sealing of all Penetrations. Boreholes into the repository system will be adequately sealed before the repository is decommissioned (495). Seals should maintain their integrity throughout, and as far beyond, the thermal period as is reasonably achievable (Section II.A.1, Objective 1) and should prevent radionuclides or toxic chemicals from reaching the biosphere in unacceptable quantities (Objective 2). The characteristics of the seal that are of primary importance in determining its adequacy are its long-term durability, its ability to prevent transmission of fluids, and its mechanical properties (496).

2. Prudent Use and Siting of Boreholes. As new boreholes are used and sited, useful means of reducing their potential impacts on repository performance will be taken into consideration. To the extent possible, work requiring surface boreholes over the proposed repository will use existing holes to minimize the total number of holes requiring sealing. Similarly, any new holes will be utilized for as many purposes as possible and will be located to coincide with proposed repository shaft or pillar locations (497). Where possible, boreholes will be located outside the immediate repository area (498). These practices in the use of boreholes will limit the number of boreholes requiring sealing and mitigate the possible impact of seal degradation either by providing a secondary barrier or by ensuring that the borehole will not provide a pathway to water through the repository (499).

3. Location and Characterization of Existing Penetrations. All reasonable efforts will be made to locate and adequately seal previously existing boreholes. The following recommended procedures (500) or an equivalent will be used:

   (a) Search existing local and state agency, oil, and service company records.

(b) Interview local residents, landowners, merchants, and service companies.

(c) Study aerial photographs for ground depressions and other possible evidence of drilling.

(d) Use magnetic-type locators for surface casings.

(e) Conduct on-site search and excavation.

4. Shaft or Borehole Construction. Final sealing will be considered in the selection of shaft designs and excavation techniques to facilitate sealing. This consideration of shaft/borehole construction will include the extent of fracturing in the surrounding rock caused by the excavation technique or residual stresses in the rock. If extensive fractures are created, it may become necessary to fill those fractures. Techniques that cause such extensive fracturing will be modified if possible. In addition, techniques used to seal aquifers during shaft construction should not preclude or reduce the effectiveness of shaft sealing (501).

II.E.2.4.3 Status of Knowledge on Penetration Sealing

Assessments of the containment and isolation provided by mined geologic repositories have not indicated that complete penetration sealing is required. The appropriate specifications for sealing will be determined from the results of consequence assessments for a particular site (502). These assessments are discussed in Section II.F.1. Nevertheless, because borehole seals will be an important redundant barrier, the Department is aggressively developing borehole sealing techniques to provide an additional measure of confidence that containment and isolation will be maintained. The function of seals will continually be reassessed as designs are developed for specific sites.

The sealing of penetrations used in oil and gas production, mining operations, and the disposal of chemical wastes and brines in deep wells has provided related experience. Successful sealing of penetrations associated with underground weapons tests has been achieved at the Nevada Test Site. These seals have withstood not only high temperatures and pressures but also strong transient ground motions without losing their integrity. The experience accumulated to date must be supplemented with further research and development, because repository seals must retain their integrity for much longer periods of time than those considered in previous applications.

The evaluation of penetration sealing has been under way and has included the following:

1. Evaluation of the materials and techniques used by oil and gas industries (503).

2. Hydrothermal transport in situ sealing (504).

3. Sealing by earth melting (505).

4. Sealing by compacted natural earth materials (506, 507).

5. Availability of in situ instrumentation (508).

6. Studies of salt dissolutioning associated with open penetrations (509-511).

The current program consists of three principal elements (512):

1. Systems analysis and consequence analysis to assess the role of seals in providing geologic containment and isolation.

2. Design, analysis, and evaluation of plug configuration materials in laboratory and field studies.

3. Evaluation of the long-term stability of sealing materials.

These efforts are directed toward the development of appropriate design specifications for seals and an evaluation of the response of various candidate seal materials in the environments that will be encountered in the various geologic media associated with a mined geologic repository.

The development of design criteria for seals will include conservative design margins so that final specifications will ensure appropriate seal longevity. The quantification of these design criteria, including a range of acceptable materials and seal geometries, is expected to be completed in late 1982. The materials studies supporting seal-design activities will continue throughout the development of the sealing techniques. Using the experimental results of field emplacements and laboratory development, prototype seal designs for repositories are to be completed in late 1982, with the design for seals for a specific site planned to be completed 2 years after a site is selected (512).

Past activities have included material studies (513-516) and the sealing of two drill holes: the AEC-1 well in Lyons, Kansas (517), and ERDA-10 (513) in southeastern New Mexico. Emplacement techniques and operational procedures were evaluated during these tests, in addition to the evaluation of seal materials. The Kansas seal was emplaced in 1973, using expanding portland cement with epoxy resin sections at the top and bottom of the salt zone. ERDA-10 was plugged in 1977 with alternate segments of grout tailored to the stratigraphy. Test specimens obtained during core drilling are now undergoing laboratory analysis. A recent report (518) has summarized the application of past experience in oil and gas exploration, mining, and deep-well disposal of chemical wastes to the sealing of a radioactive waste repository and ongoing Department activities (518).

Laboratory and field programs currently in progress are evaluating the properties of several candidate materials (519) over a range of environmental conditions. These studies include the examination of factors affecting the geochemical longevity of seals (520, 521). In situ studies in bedded-salt (522) have included the recovery and laboratory evaluation of cement seals used in potash-exploration drill holes. Observations revealed that no serious degradation had occurred after 17 years of curing at a depth of approximately 1,000 ft and that permeabilities were in the millidarcy range (523). X-Ray diffraction patterns and scanning electron microscope analysis of the 17-year-old seal compared with those of the ERDA-10 grout at 2 weeks and 1 year showed similarities in composition and microstructure. There was evidence of relatively little exchange or reaction between the cement plug and the surrounding rock over the 17-year period (524).

A field test currently in progress has evaluated the in situ permeability of a cement seal that was emplaced at a depth of 4,500 ft in anhydrite rock underlying a bedded-salt deposit in New Mexico (525). In this test, a seal 8 ft long was used to seal off an 1,800-psi aquifer. An effective liquid permeability of approximately 50 microdarcies has been observed in a series of measurements over a period of 3 months (526). Testing of this seal and other seals in the same borehole over the next few years will be continued.

Investigations will continue to develop various materials to operate in the environments that seals will encounter. These will include the geochemical characterization of specific sites and the evaluation of materials under the same conditions. In addition, techniques for efficient seal emplacement methods, quality assurance techniques, and in situ characterization of seals will be developed. It should also be noted that significant advances in the sealing technology are expected before final sealing during repository decommissioning is required.

II.E.2.5 Backfilling

One of the measures that can be taken to reduce long-term stresses and hasten the return of the geologic structure to its original undisturbed state is backfilling the underground excavations; i.e., refilling mined-out areas after waste emplacement operations have ceased. Backfilling in ordinary mining operations has a long history.

II.E.2.5.1 Potential Impacts of Backfilling

The impacts of backfilling on the long-term containment isolation capability of the geologic systems may include:

1. Reducing the stresses and strains associated with the ultimate reconsolidation of the geologic structure.

2. Reducing the accessibility of the radioactive materials to transport agents such as ground water and man.

3. Retarding the potential rate of radionuclide transport by ground water by restoring and/or enchancing the radionuclide-retardation capability of the geologic material.

II.E.2.5.2 Desirable Backfill Characteristics

Backfilling is one of the design features proposed to mitigate the effects of excavation on the geologic formation and possibly enhance its long-term containment and isolation capability. Potentially desirable properties for backfill material (527) include:

1. Low permeability.
2. High radionuclide-retardation potential.
3. Low solubility.
4. Expansion potential.
5. Chemical stability.
6. High thermal conductivity.
7. Thermal stability.

It is not expected that any one backfill material will have all these characteristics.

II.E.2.5.3 Status of Knowledge on Backfilling

As noted in the introduction, the backfilling of underground excavations is a historic practice. There is, therefore, a considerable body of practical knowledge and experience regarding methods of backfilling.

Procedures and equipment for backfilling operations are widely available, and their application to repository operations is primarily a matter of good engineering (528). For example, the method selected in a recent conceptual design (106) envisions the use of conveyors to transport the tailings material directly from the working faces of the excavation to areas scheduled for backfilling. As it arrives, the tailings salt would be piled in the old excavation by a centrifugal thrower capable of piling the material to within 2 ft of the ceiling. The density of backfill emplaced in this manner has been estimated (528) to be about 60% of theoretical (527).

Backfilling is planned for all the repository conceptual designs completed to date (529-532) after a careful identification and consideration of all the effects that backfilling may have on the operation and decommissioning of the repository and on the long-term containment and isolation of the waste.

A potential disadvantage of backfilling in the short term is a local increase in temperature. Studies (527, 528) indicate that the backfilling of filled repository areas will result in temperature increases at points near the storage room (528). The two major causes of these temperature are the relatively poor thermal conduction properties of the uncompacted backfill material and the elimination of ventilation air flow through the backfilled storage areas. The predicted temperature increases are very local. The thermal history of the overall geologic structure does not appear to be significantly affected by backfilling. Therefore the increase in temperature due to backfilling is not expected to significantly impact the structural stability of the repository in either the short or the long term (528, 533).

II.E.2.6 Repository Structure Summary

Potential adverse impacts of constructing and operating a repository will be limited by application of the rational approach to design as described in this section. This approach is based on an understanding of the phenomena that affect the performance of the repository and of the measures that can be taken to minimize the potential for any adverse impacts due to heat production by the waste. Radiation effects on the repository are expected to be insignificant because they are restricted to the very near field.

The measures taken to minimize the impacts of constructing and operating a repository can be summarized as follows:

1. The impacts of the repository excavation on structural stability will be limited by using low extraction ratios, by using excavation techniques that are highly developed and widely applied, and by backfilling the rooms and tunnels.

2. The thermal impacts will be minimized by limiting the thermal loading and thus the temperatures in the repository.

3. The migration of radionuclides from the waste emplacement are will be restricted by use of sorptive backfill materials that are being developed and tested.

4. The penetrations into the repository will be sealed. Progress in the development and testing of the requisite sealing materials and techniques has resulted from current research programs. Programs for seal development are expected to result in satisfactory designs and materials in time for repository closure.

II.E.3 Protective Measures Against Human Intrusion

II.E.3.1 Introduction

The successful isolation of high-level radioactive waste from the biosphere over the long periods of time requires that the waste be unaffected by natural events and processes and also be satisfactorily independent of future activities of man. Much consideration has been given to the concept that future human societies may either willingly or unknowingly engage in activities that could compromise the effectiveness of a waste disposal system (534, 535). Concerns have arisen about the state of technological advancement or regression that could be expected of future generations; of the potential effectiveness of long-term control measures to prevent future human intrusion into the disposal system; of the need to site waste disposal systems in areas where future exploration for deep resources would be unlikely to take place in a manner that would unintentionally breach the repository system; and of the need to dispose of waste in a manner that would allow recovery by future societies, if so desired.

It is a basic premise in this Statement that, although this generation bears the responsibility for protecting future societies from the waste that it creates, future societies must assume the responsibility for any risks which arise from deliberate and informed acts which they choose to perform.

At issue is the protection of the public health and safety from waste releases unintentionally initiated by future human activities. The complete prevention of human-induced releases is desirable but probably not reasonable. Reasonable objectives would be, however, to (i) reduce the likelihood of human-induced releases, and (ii) mitigate the consequences of human-induced releases.

The role of the natural systems (II.D), the waste package (II.E.1), and the repository structure (II.E.2) have been discussed relative to the containment and isolation of wastes. This section describes the measures being explored by the Department to implement the two objectives stated above relative to human intrusion. The focus is on the prevention and mitigation of unintentional releases from waste disposal systems by persons/societies who are unaware of a repository's existence. The categories of future human activities that require consideration relative to their potential for human-induced repository releases are discussed along with possible preventive and mitigative measures under consideration in the NWTS Program.

The discussions in this section are general and are primarily intended to indicate the consideration being given to protecting against such releases, beyond what is evident through the NWTS Program siting criteria (536). Research has been conducted and the systems discussed in II.D and II.E.1 and II.E.2 can provide the desired mitigation should intrusion take place, but considerable additional study is required to fully develop methods to protect against the occurrence of human-induced releases.

II.E.3.2 Categories of Future Human Activities Considered

Several categories of future human activities have been suggested as being potentially disruptive of a waste disposal system's effectiveness (534, 535, 537). In general these activities are as follows:*

1. Exploration or excavation for resources.
2. Alternative land/host rock use.
3. War or sabotage.

Exploration and excavation would include exploratory drilling, conventional mining, solution mining, and other activities which could physically breach the host rock and lead to pathways (e.g., establish hydraulic continuity) or provide mechanisms (e.g., mining of contaminated host rock) for waste release to the biosphere. Scenarios have been postulated for solution mining of salt (537, 538) in which the lack of multiple redundant barriers in the reference calculation resulted in unacceptable effects. They also point out the desirability of providing measures to reduce the possibility of future human intrusion. Such measures, as discussed in this section, would provide protection against a variety of potential human-induced release mechanisms.

Alternative land or host rock use would include the construction of dams (of concern to ground water flow characteristics) (539), attempts to utilize the host rock for the storage of other materials (e.g., petroleum or gas storage, storage of other hazardous wastes), and any other activities involving land or host rock usage that could decrease the effectiveness of the host rock or hydrologic systems. Preventive and mitigative mechanisms for alternative land or host rock use are discussed later in this section.

Effects of war refers to adverse impacts resulting from the detonation of powerful explosive devices. It has been postulated that the explosion of nuclear weapons could provide a direct path of exposure from the

*The categories given are indicative, but not necessarily all-inclusive, of activities that may require consideration.

repository to the environment or otherwise decrease the effectiveness of the repository system. Cratering depths determined by using nuclear explosives indicate that the direct exposure of wastes emplaced in a repository would require the detonation of a device many times larger than any that has been tested. Detonations of considerable size near a repository might affect the local hydrologic system; however, the direct effects of the weapons are significant in their own right and tend to mask any subsequent repository effects. The depth and design (see II.E.2) of the repository system and the nature of repository releases (long lasting, low-level releases of radioactive materials, Objective 2 in II.A.1) compared to direct weapons effects would make the repository both extremely difficult to breach and undesirable as a military target. Sabotage, a corollary consideration, could be directed against a waste disposal system; nevertheless, as with the effects of war, the depth and design of a deep geologic repository would not render it an attractive or effective target for sabotage.

II.E.3.3 Protective Measures

A widely discussed protective measure to lower the likelihood of human-induced releases involves the selection of the repository site. As indicated in the site selection criteria (536) for the NWTS Program (see II.D.3), care will be exercised in choosing sites to minimize the likelihood of human activities directed toward gaining access to natural resources and future land-use conflicts (e.g., dam construction). Although future societies may actively seek materials that are not now regarded as significant resources, the likelihood of their needing to recover resources from a repository site can be controlled to some extent by careful consideration of such factors during site selection.

Beyond site selection factors, additional protective measures will be used to communicate knowledge of the existence of repositories to future generations and to mitigate the effects of unintended human-induced losses of containment, if they were to occur. These protective measures are indicated in Figure II-23 and are discussed below.

II.E.3.3.1 Institutional Controls

Although active institutional control mechanisms can be highly effective, the duration of their long-term effectiveness is questionable (536). The long-term use of active institutional controls is inconsistent with Objective 6 in II.A.1. Examples of active institutional controls include land-use control rights by the Federal Government; the requirement for permits to perform certain types of operations near the repository site; the use of fences or other physical barriers to limit access to the site; and/or the employment of "caretakers" such as a park police force or security guards, charged with controlling access to the site. The use of active institutional controls is dependent upon the continued existence of institutions charged with the responsibility for maintaining the repository site. The continued existence of institutions charged with that responsibility cannot be ensured over the extended period of time required for waste isolation (540). Long-term active institutional controls will not be assumed in the NWTS Program safety analyses.

II.E.3.3.2 Public Awareness

The second protective measure against human induced release is that, regardless of the existence of an institutional structure to actively control access to the repository site, local public knowledge of the existence of a repository is likely to be carried forward in time because of its importance--i.e., due to the high degree of perceived risk associated with the disposal of radioactive waste.

Information that affects health and safety has been known to transcend generations. Sometimes an original cause and effect is transformed into institutionalized prescriptions and proscriptions. For example, Hebrew and Islamic laws still proscribe eating pork, even though the principal benefit of instituting such laws--likely the prevention of trichinosis resulting from the consumption of meat from infected swine--is no longer a significant health concern. As an indication of the longevity of such awareness, even

PURPOSE OF CONTROL		NATURE OF CONTROL
POSITIVE PROTECTION FOR DURATION OF STABLE GOVERNMENTS.	ACTIVE INSTITUTIONAL CONTROLS	INSTITUTIONAL MANAGEMENT OF ACTIVITIES WITHIN SITE BOUNDARIES
LONG-LASTING INDICATOR OF REPOSITORY EXISTENCE NOT UNIQUELY TIED TO REPOSITORY SITE; AND LONG-LASTING INDICATOR OF REPOSITORY EXISTENCE NOT TIED TO GOVERNMENT STABILITY	PUBLIC AWARENESS	ACCESSIBLE PUBLIC RECORDS, LIBRARY/TOWN HALL RECORDS, TIME CAPSULES, INTERNATIONAL & NATIONAL ARCHIVES; AND SOCIETAL MEMORY, AND WORD OF MOUTH/LEGEND, PASSING ON OF INFORMATION RELATIVE TO POTENTIAL DANGERS
VISIBLE ON-SITE WARNING OF REPOSITORY EXISTENCE	PERMANENT MARKERS	VARIETY OF HIGHLY VISIBLE HIGHLY DURABLE MARKERS CHOSEN TO ENSURE LONG SURVIVAL OF ONE OR MORE MARKERS
INDICATION OF REPOSITORY DUE TO OFF-NORMAL CONDITIONS THAT WOULD BE APPARENT TO FUTURE INVESTIGATORS AND WOULD CURTAIL FURTHER ACTIONS	WASTE ANNUNCIATION	PHYSICAL AND CHEMICAL TRACERS IMPLANTED IN HOST ROCK AND SYSTEM HAVING A LIFETIME UNTIL INTRUSION, LONG-POST-INTRUSION LIFETIME, THERMAL/RADIATION SIGNATURES
ENGINEERED MEASURES TO PROTECT SAFETY OF FUTURE EXPLORERS AND PUBLIC DURING AND FOR SOME TIME AFTER EXPLORATION OF SYSTEM	ENGINEERED BARRIERS	PHYSICAL AND CHEMICAL BARRIERS DESIGNED TO EXIST THROUGHOUT INTRUSION AND FOR LONG TIME THEREAFTER. ESSENTIALLY A PRE-INTRUSION LIFE EXPECTANCY.
SYSTEM PERFORMANCE CHARACTERISTICS IS DESIGNED TO PROTECT PUBLIC IF SYSTEM INTEGRITY IS DECREASED BY ANY MECHANISM	NEAR FIELD & FAR FIELD SYSTEM BARRIERS	REPOSITORY SYSTEM CHOSEN TO MITIGATE RELEASES TO BIOSPHERE REGARDLESS OF INITIATING EVENT, I.E., NATURAL OR HUMAN INDUCED

Figure II-23. General Levels of Defense Against Future Human Activities when shifts in societal sophistication eliminate the source of concern, institutional proscriptions frequently are left intact, as the above example illustrates. The two primary mechanisms for continuance of public awareness are these:

1. The existence of public records in permanent form related to the repository and placed in a wide variety of locations.

2. Societal memory, i.e., passing on, by word of mouth, information about potential dangers associated with the repository.

The existence of permanent records widely disseminated throughout all countries of the world and indicating the exact location of repository sites, made available at public libraries, government centers, computerized information search centers, and in time capsules, should reduce the probability of all records of the repository being lost. It is likely that global communications abilities will continue to grow in sophistication and, even given the potential changes in governments, records of the existence of repositories would be preserved and perpetuated by appropriate societal mechanisms.

There are many examples of records that have been passed on from generation to generation for centuries. For instance, the various religious documents belonging to the major religions of the world fall into that category. Ancient Greek writings on philosophy and mathematics, as well as Greek masterpieces in literature, go back in history over 2,000 years. During the past century, many written records have been discovered during archaeological excavations. These include cuneiform tablets belonging to the ancient Assyrian culture and Egyptian hieroglyphics. Both are thousands of years old. Literary writings from many cultures have been preserved, either through translations or through copies of original manuscripts kept in libraries throughout the civilized world. A more closely related example consists of external records describing mining activities in the Harz Mountains of Germany; these records have been preserved for more than 900 years and are still usable despite rather drastic changes in technology, government, and social structures. The above examples illustrate that literate societies can and do preserve information and that their information is passed on to future generations through written records. In nonliterate societies, myths, tales, and legends are often of great functional importance serving as an important means of transmitting information from one generation to the next.

Notwithstanding the continued existence of permanent records, societal memory--the passing down from generation to generation information that is considered to be important to the general health and welfare--is a human trait. Many examples from history testify to that trait. People living in rural areas have transferred by word of mouth for generations what types of mushrooms, berries, herbs, etc., are safe to eat. Knowledge of the medicinal properties of many plants, chemical substances, spring water, etc., has also been passed on from generation to generation (541).

II.E.3.3.3 Permanent Markers Indicating the Existence of the Repository

Surface monuments and markers designed specifically to warn future generations of the repository location should be reasonably effective. The use of such markers has been recommended by the Environmental Protection Agency in its proposed criteria (540).

Human history is replete with examples of works of man that have stood for millennia reflecting the beliefs and concerns of bygone generations. Examples include the pyramids of Egypt (built approximately 4,800 years ago) and the Mayan temples; Stonehenge and other remnants of early culture; cities and civil structures from Greek and Roman times; temples in Southeast Asia; drawings made on the Earth's surface by South American civilizations; prehistoric cave drawings on the ceiling of the Altamira caves in Spain (in excess of 10,000 yrs old), and those in France and in the United States; and earthworks and dwellings created by North American Indian cultures. Present investigators may not fully understand many artifacts of the past, but scientific investigation into the meaning or significance of such phenomena is usually rigorous (542).

Given the technical sophistication of the present society relative to the ancient civilizations, it is reasonable to assume that a series of diverse permanent markers, each designed to withstand several millennia of natural and human activities, could ensure continued awareness of the existence of the repository system, or at least indicate the existence of an unusual and possibly hazardous condition. The development of permanent markers will be pursued in the NWTS Program.

II.E.3.3.4 In Situ Indicators

Concern has been raised that if a repository site were to be violated for future industrial purposes (e.g., solution mining of salt) (537), the developers would not be aware of the existence of the radioactive waste without performing special monitoring or radiochemical analyses. In principle, in situ "tell-tales" could be placed in the repository media in such a manner that they would alert future explorers to the existence of the repository due to "unique" easily detectable conditions existing in the host rock. The purpose of the tell-tales would be to re-alert society to the existence of the repository if the preceding measures were to prove unsuccessful at some point in the future. Further study is needed to determine the most appropriate measures that should be used to effectively carry out this function.

II.E.3.3.5 Engineered Barriers

Engineered barriers present in the repository to mitigate the impacts of potential natural intrusion events or processes also would be effective in mitigating impacts of potential human-induced release. For example, long-lived waste packages (II.E.1) and repository design features (II.E.2) would retard radionuclide transport for both human and naturally induced ground-water intrusions. Additional site-specific barriers will be considered for their effectiveness in mitigating the consequences of potential releases.

II.E.3.3.6 Natural Isolation Barriers

Analagous to the rationale for engineered barriers, the hydrologic and geologic features of the repository (see II.D.1) would be effective for mitigating either natural or human-induced intrusions into containment.

II.E.3.4 Human Intrusion Summary

The six protective measures indicated in Figure II-23 and discussed herein will assist in providing assurance that future human activities will not inadvertently disrupt the integrity of the repository system in a manner which would be of concern to the public health and safety. The first three measures are directed toward perpetuating the knowledge of the existence of the repository system to avoid human-induced events. Since, as noted by the Environmental Protection Agency (540), it is difficult to predict the length of time that active institutional controls could be relied upon, credit for the first measure will not be assumed in NWTS Program evaluations. Societal memory, accessible public records, and permanent markers could nevertheless provide a longstanding indication of the existence of the repository, independent of institutional controls.

The above mentioned measures could reduce the likelihood of future human activities causing unknowing and/or unwilling intrusion of the repository. Beyond those measures, the fourth measure, in situ indicators, would increase the likelihood that a repository would be rediscovered, and intrusive events would cease should the effectiveness of the above measures diminish with time. Knowledge of the repository existence could thereby be restored to society's memory. The first four measures could, in a self-perpetuating fashion, continue to remind society of the existence of the repository system, thus continuously maintaining a low likelihood of inadvertent human intrusions into the repository.

The fifth and sixth measures rely on the basic natural and man-made barriers discussed in II.D and II.E, respectively. Those barriers would minimize the impact of any loss of repository integrity, be it through natural or human causes.

It is reasonable to conclude that (i) the likelihood of future human activities of a nature which could adversely affect the integrity of the repository can be reduced to an acceptably low probability through the use of appropriate protective measures; and (ii) the impact of any such future activities, were they to occur, could be adequately mitigated by the multiple natural and man-made barriers included in waste disposal systems.

II.F PERFORMANCE ASSESSMENT OF MINED GEOLOGIC DISPOSAL SYSTEM

Preceding chapters of Part II describe the natural (II.D) and man-made (II.E) parts of the mined geologic waste disposal system. This chapter details the methods of analyzing the system to determine whether it can be expected to meet the performance objectives and requirements stated in Chapters II.A, II.D, and II.E. The description is in four sections.

Section II.F.1 describes the methods that have been developed for analyzing the performance of the disposal system after the waste has been emplaced and the repository has been scaled. During the long time for which the waste will remain hazardous, many phenomena might affect the disposal system: natural events and processes, human actions, and impacts exerted by the waste and the repository. The performance analysis must predict the combined effects of these phenomena. If the analysis identifies any effects that could release the radionuclides and deliver radiation doses to people, it must estimate the magnitude of those radiation doses.

These requirements on the long-term analysis present unusual problems that have not had to be solved in analyses of other systems. Furthermore, the disposal system itself contains components that are different from those of previously analyzed systems. For these reasons, the assessment of long-term performance has required the development of special methods. To make the needed predictions, these special methods use mathematical descriptions, called models, of the phenomena that might affect the system. Section II.F.1 describes how those models are developed and refined; it explains how they are applied to disposal systems to determine whether the systems will retain the radionuclides effectively, in accordance with Objectives 1 and 2 of Chapter II.A.

Section II.F.2 describes the role of experiments and observations of natural phenomena in formulating, developing, and verifying the models. These experiments necessarily involve laboratory and field, or in situ, tests. The section also summarizes the NWTS programs that are providing the required data.

Section II.F.3 describes the methods for assessing the performance of the repository during the operational phase, before it is closed.

This assessment generally does not require the development of special methods because the operations are similar to those in other common systems, especially the operations carried out routinely in the nuclear fuel cycle. However, the heat generated by the waste creates some effects in the host rock that present some unusual engineering problems; these problems are examined by the same methods used for the long-term assessments.

Sections II.F.1 and II.F.3 are primarily concerned with methods for predicting the potential environmental effects of releases of radiation, if any, during the long-term and the operational phases of geologic disposal. In contrast, Section II.F.4 is primarily concerned with methods for predicting the potential environmental effects of a repository during construction, operation, and long-term containment and isolation. No special methods need to be developed for these predictions beyond those already described in Section II.F.1 for the impacts of radiation. The section presents an example listing the environmental impacts of a disposal system, including the impacts of radiation.

II.F.1 Assessment of Long-Term Performance

Predictions of the long-term performance of the disposal system after the waste has been emplaced and the repository closed are important to site selection, repository design, and waste-package design. The technical basis for deciding whether a disposal system meets established performance criteria will be a comparison between those predictions and the criteria. This section describes the general NWTS approach to long-term performance-assessment. It describes the status of the development, verification, and application of the performance-assessment methods (543).

An assessment of the long-term performance of a repository analyzes the phenomena that might release radionuclides from the waste and the phenomena that might move radionuclides into the biosphere to people. These phenomena may be roughly classified as those that occur in the "near field" close to the waste and those that occur in the "far field" at greater distances from the waste--that is, in the host rock and beyond. Although these two regions are not separated by a precisely defined boundary, the distinction is useful because the heat and radiation from the waste are more intense in the near field. Different methods of analysis are therefore appropriate. Near-field analysis primarily studies the effects of heat, radiation, and the construction of the repository on the waste package and the repository. Far-field analysis primarily studies the effects of phenomena that arise from the head produced by the waste, from natural phenomena, and from human actions. These phenomena usually appear outside the repository, in the geosphere and the biosphere. Near-field and far-field performance must both be considered in determining how well the natural and the man-made portions of the disposal system meet the performance objectives and requirements described in Chapters II.A, II.D, and II.E.

Subsection II.F.1.1 describes in general terms the approach used to predict long-term performance. The next two subsections are more detailed: Subsection II.F.1.2 describes the development and verification of the methods for assessing far-field performance, and Subsection II.F.1.3 describes similar methods for near-field performance. Subsection II.F.1.4 briefly describes the results of studies that have used the performance-assessment techniques and summarizes several specific examples. Subsection II.F.1.5 draws on the preceding subsections to present conclusions about the status of the techniques and about the results of the studies that have used them.

II.F.1.1 General Approach to the Assessment of Long-Term Performance

As summarized in Figure II-24, the assessment of long-term performance treats four principal components of the complete waste-disposal system: the waste package, the repository structure, the geosphere, and the biosphere. The geosphere and the biosphere, which are of primary importance to the far-field analysis, are also referred to as the site and the environs of the site.

The analysis begins by defining the state of the system for each principal component at the time the repository is closed. The analysis then describes the effects of phenomena that might breach the repository and release radionuclides from it; for this description the analysis uses the mathematical methods described later in this section. These methods predict the future state of the system components, the manner in which transport agents like ground water may move into the system to reach the waste, and the manner in which nuclides may move through the system to reach people. Finally, the analysis predicts the potential effects of the radionuclides on the people whom they are assumed to reach. As shown by the horizontal arrows in Figure II-24, all of these analyses are strongly interrelated and must be considered together in predicting the performance of all or any of the components of the disposal system.

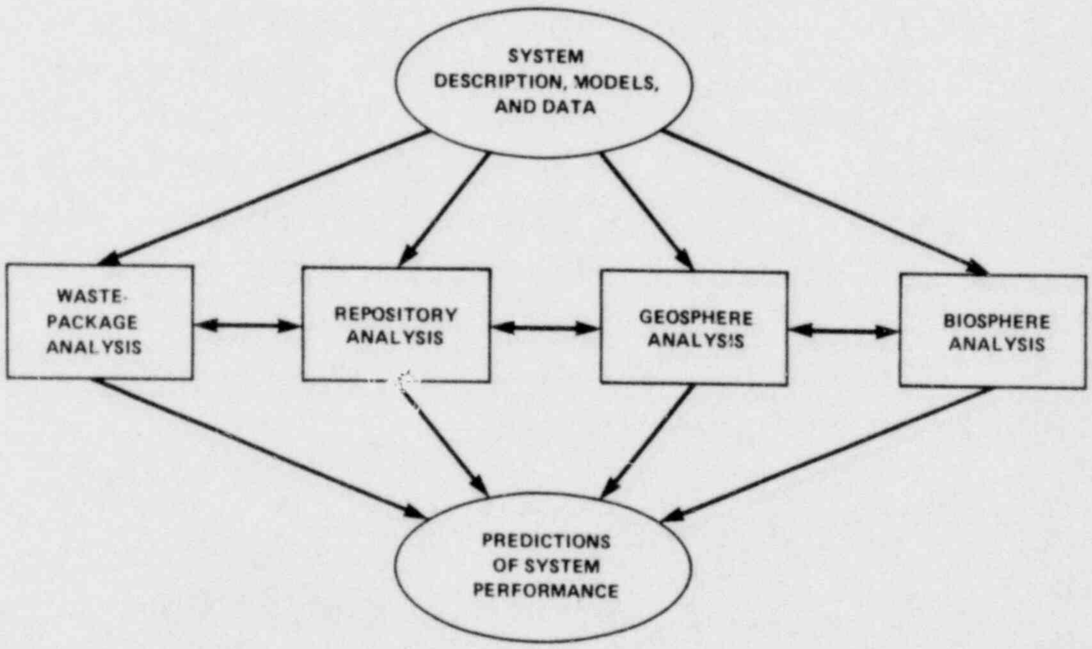


Figure II-24. General Approach to Performance Assessment

In order to make quantitative predictions, analyses like these require the use of mathematical descriptions, called models, of the phenomena. Before the models can be used with confidence, they must be developed and verified. The models contain parameters whose numerical values must be specified in order to describe the system and the phenomena; the specification of these values requires experimental data or theoretical calculations. Using appropriate models with well-specified parameters, the analysis predicts the long-term performance of proposed repository systems. The results can be applied to the decisions that are to be made in the NWTS Program. Sensitivity and uncertainty analyses, defined and discussed below, are important parts of this general approach.

The use of the term "model" may cause some confusion because it is used with somewhat different meanings in different parts of the scientific community working in radioactive-waste management. Geologists, for example, often use the word to mean a description of the Earth's structure beneath its surface; this description is not necessarily in mathematical terms. Other workers in waste management use the phrase "conceptual models" to mean non-mathematical descriptions of phenomena. In this discussion, however, models are descriptions expressed by a set of mathematical relationships that may be either simple or complex. It is these descriptions that produce quantitative predictions of the long-term performance of a repository.

The next four sections discuss in more detail the general approach to performance assessment.

II.F.1.1.1 Model Development and Verification

The development and verification of individual performance-assessment models follow a logical procedure. First, the phenomena to be modeled must be identified. Then, available information about those phenomena and about the disposal-system is collected and organized. This information comes from the observation of natural behavior, laboratory experimentation, and field testing. Next, models based on this understanding are developed. These models can be empirical (i.e., based on direct use of the observations) or theoretical (i.e., based on interpretation of the observations according to known physical laws). The predictions of the models are then compared with additional observations of the disposal-system phenomena under conditions different from those used for developing the models. If the comparison shows significant differences between predictions and observations, the models are modified and tested again. The iterative process of model modification and verification testing continues until a satisfactory agreement between prediction and observation is obtained. Scientific peer review of both the development and the verification is important in determining the acceptability of the models. The role of experimentation in supporting this process is discussed further in Section II.F.2. Although the process is followed for each model, it is necessarily limited for models of phenomena that take place only over extremely long times or are of such low probability that they cannot reasonably be tested.

This development of models begins with simple models, called "individual" models, of specific phenomena. Then, more complexity is added to describe additional phenomena; some initially considered phenomena may be omitted from a model if experiments show them to be of minor importance. The more complex models that describe several competing phenomena are called "coupled" models. The continued testing of a model during the development process builds confidence in the model and is an important part of model verification.

In their ultimate development, the individual and coupled models will be combined to form a system in which an executive model manages the interactions between them. Such a complete model is not expected to be available until 1983.

II.F.1.1.2 Data Requirements

Specifying the parameters in the models requires information on all four principal components of the waste-disposal system. The waste package is important to meeting Objective 1 (II.A.1), containment, and the geosphere to meeting Objective 2 (II.A.1), isolation. The structure of the repository is important to both containment and isolation.

The information on the waste package describes the characteristics of the emplaced spent fuel and the heat and radiation it generates. Examples of the required data are radionuclide inventories, canister wall thickness and corrosion rate, and the rate of release of radionuclides.

The required information on repository structures includes design specifications of the mined repository and of the engineered barriers built into it. It also includes quantitative descriptions of the response of the repository system to the phenomena that may affect it. Some of the required data are the distance between emplaced packages, the permeability and chemical resistance of the engineered barriers, and the thermal and mechanical properties of the host rock.

The required information on the geosphere includes the description of the rocks and hydrologic systems in the region around the site. Examples of the required data are the detailed stratigraphy of the region, the transmissivities of any aquifers, and the sorption characteristics of the aquifer rocks.

The needed information on the biosphere includes descriptions of the ecosystem at the site and of the biological pathways through which radionuclides might move to people. Some of the required data are the food chains in the region and the rates at which water is consumed by people and animals there. Because most of these data are specific to a particular site, the performance-assessment data base will become more complete as the investigation and the development of each site proceed.

II.F.1.1.3 Uses for Assessments of Long-Term Performance

The methods for assessing long-term performance are important to analyses of safety and environmental impacts. Such analyses have four general applications in the NWTS Program: (i) the evaluation and selection of disposal sites, (ii) the development and evaluation of designs for the mined part of a repository, (iii) the design and evaluation of engineered barriers, and (iv) the qualification and licensing of the disposal system. Each of these applications supports important decisions to be made in the development of systems for the disposal of radioactive waste.

The choices among candidate sites and repository designs will require a comparative evaluation of the abilities of the host rocks and the surrounding formations to prevent and retard the release and transport of radionuclides. Such an evaluation will use the appropriate assessment models together with data from the site-exploration activities and from the proposed design of the repository, including the design of the engineered barriers.

The engineered barriers in a repository will include the waste-package systems, the backfill used in the disposal rooms, and the seals in shafts and boreholes that penetrate the repository (see II.E.1 and II.E.2). The choice among candidate engineered barriers will require a comparative evaluation of the expected performances of the barriers individually and as a set. This evaluation will use the assessment models--primarily the near-field models--together with data from the tests of the barriers, from the proposed repository design, and from site studies.

The qualification and licensing of a disposal system will require a demonstration that the expected performance of the entire system is acceptable. This demonstration will use both the near-field and the far-field assessment models and the entire set of data describing the system, including the results of in situ tests.

In making the calculations necessary to support decisions in the above four areas, the Department will use the following approach:

1. Early in the development of a disposal system, when detailed information about the specific system of interest is not yet available, bounding performance calculations will be made using the models and reasonable but conservative values for the model parameters. These bounding calculations will define upper limits to the hazards that the disposal system could produce.

2. Later in the development, when detailed information about the specific system is available, multiple analyses of system performance will be made using the models and the entire distribution of values for the model parameters. These calculations will provide realistic predictions of system performance along with an overall measure of the uncertainty associated with those predictions.

II.F.1.1.4 Sensitivity and Uncertainty Analyses

The predictions made with the models used in the performance assessment are subject to some uncertainty; each prediction lies in a range of possible values. This uncertainty arises from three major sources: uncertainty in the ability of the model to describe adequately the phenomena it portrays, uncertainty in the data used to specify the parameters in the model, and uncertainty in the assumptions about conditions on Earth in the far future. The uncertainties in the data may themselves arise from two sources: a lack of precision in the measurements and the inability of experiments or observations to precisely measure all the needed geologic data without destroying the rock system in the process.

In general, the uncertainties involved in assessing the long-term performance of a disposal system tend to be larger than the uncertainties involved in most other systems analyses. Few other systems require analyses extending so far into the future, and few include components that, like the host rock, are so difficult to test non-destructively. To increase confidence in the predictions of the models, it is therefore necessary to estimate the overall uncertainty in the predictions. Uncertainty analysis is a tool for performing this task.

The uncertainty analysis of a model studies the uncertainties arising from each of the sources. It determines their effect on the uncertainty associated with the predictions made with the model. It also predicts the "residual uncertainties" associated with the models--uncertainties that would be expected to remain after the model has been completely developed and verified and after all the required data have been collected. In other words, uncertainty analysis combines all the uncertainties associated with the parts of a model and determines the composite uncertainty associated with the predictions of the model.

Another tool that is useful in performance assessment is sensitivity analysis. Some parameters in a model have a greater effect on the predictions than other parameters. Sensitivity analysis evaluates the effects of the parameters by identifying the parameters whose variation has the greatest effect on the predictions.

Sensitivity and uncertainty analyses are being applied to the models to establish priorities for research that will help develop the assessment techniques. In this use, these analyses will determine the important phenomena and model parameters. The analyses will also be useful in refining and interpreting the assessments outlined in the preceding section. A major use for uncertainty and sensitivity analyses is in the design and operation of a disposal system; they will determine where additional barriers are needed and how to make the best use of design and operating margins (Objective 5). This knowledge will contribute to a logical and defensible safety analysis for licensing.

II.F.1.2 Methods for Assessing Far-Field Performance

This subsection describes the methods available for assessing the long-term performance of a repository in the far field. Models have been developed and tested for use with the techniques described in general terms in the preceding section. Although there are significant differences in the states of their development and verification (544-549), the far-field models are sufficiently advanced to be used in assessments of repositories at generic and specific sites.

During the long time in which the waste will remain hazardous, many phenomena might affect the disposal system: natural events and processes, human actions, and the impacts exerted by the waste and the repository. The performance analysis must predict the combined effects of these phenomena. Three kinds of future conditions of the system can then be identified: (i) conditions of complete containment, under which no radionuclides are released; (ii) conditions of complete isolation, under which radionuclides are released from the waste package but no people receive radiation doses; and (iii) conditions under which released radionuclides deliver radiation doses to people. In order to ascertain that Objective 2 in Section II.A.1 has been achieved, the third kind of condition must be analyzed further to estimate the radiation doses that the people might receive. The remainder of this section describes the procedure by which the conditions that might deliver radiation doses are analyzed. The discussion moves through the principal steps in the procedure, describing the far-field models that are used in the analyses.

Figure II-25 shows the procedure for calculating the far-field effects of a possible repository breach. The rest of this subsection describes the procedure in terms of the steps shown as boxes in that figure.

The box labeled "source term" describes the waste at the time the breach occurs; it specifies the radionuclides present and the physical and chemical conditions of the waste. The radionuclide concentrations at the time of the breach can be calculated from the original concentrations in the waste. Some of the physical and chemical information required for this source term can be predicted from studies of the near-field interactions discussed later in Subsection II.F.1.3. When such information is not available or is uncertain, the analysis assumes conservative conditions for the waste, in accordance with Objective 5.

The details of the analysis depend on the natural phenomena or human actions that might breach the repository; some natural phenomena typical of those that have been studied (544, 545, 548, 550-552) are climatic changes, glaciation, deformation of the host rock, meteorite impacts, and earthquakes. The analysis also depends on the construction of a scenario--an assumed sequence of events--that starts with a breaching event and includes processes that bring ground water to the waste and that move dissolved radionuclides through a path to the biosphere. A mathematical description of the scenario--shown as a block in Figure II-25--is used to predict the movement of radionuclides away from the repository.

The construction of scenarios for the long-term analysis of a repository rests on the properties of the repository and its site; the phenomena that may be possible at one site may not be possible at another. In practice, scenarios are constructed with the aid of such techniques as fault-tree analysis, event-tree analysis, predictions made by deterministic models that describe bounding phenomena, and discussions among persons with expert knowledge of the repository site and the possible breaching phenomena. A list of scenarios for the analysis of a repository will include the phenomena that are credible at the site. It may also include bounding scenarios, i.e., highly improbable scenarios whose consequences would be upper limits to the consequences of any of the more credible scenarios.

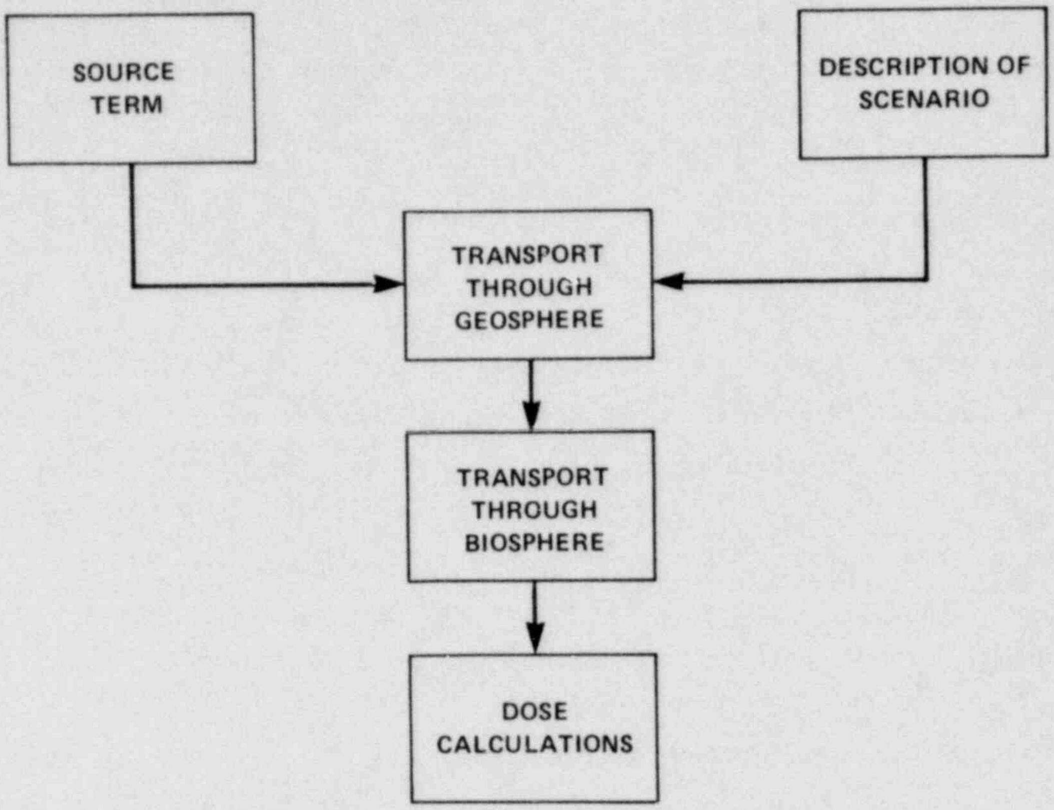


Figure II-25. Elements of Far-Field Performance Assessment Modeling

Some models of assumed far-field scenarios involving both natural and man-caused breaching phenomena have been reported (544, 545, 548, 552, 553) and have received peer review by the geoscience community. Further work on both of these types of models is planned as part of the Department's Waste Isolation Performance Assessment Program (554).

After the scenarios that need to be modeled have been identified, the next step in a performance assessment is the prediction of their consequences. Whether the scenarios will actually occur cannot be predicted with complete certainty, although their occurrence may often be characterized by probabilities. Probabilities are, however, highly uncertain for the low-probability events that have occurred in the region around a repository site only a few times in geologic history or may never have occurred there at all. For this reason, the assessment of repository performance relies heavily on predictions of the consequences of scenarios rather than on predictions of their probabilities. In licensing procedures, evaluations of probabilities primarily aid in deciding which of the many imaginable scenarios are both credible and limiting. These are the scenarios to be analyzed in detail for licensing, which is expected to rely on predictions of the consequences of limiting scenarios.

To predict these consequences, the performance analysis moves to the box labeled "transport through geosphere" in Figure II-25. This step is essentially the modeling of fluid flow into and away from the repository; ground water flow has been recognized as the principal means for transporting radionuclides from a breached repository. Tables II-11 and II-12 list some commonly used models (also called "codes" when they are programmed, or coded, for a computer) that predict transport through the geosphere; the following paragraphs explain in more detail the capabilities of these codes.

Mathematical models of fluid flow through porous media have been well developed over the past 25 years (Table II-11). Some available computer codes, such as the code entitled PATHS (555), are two-dimensional: they can predict flow along both the length and the width of an aquifer. Some two-dimensional codes, such as VTT (556), can describe more than one aquifer in a single model. Other codes, such as SWIFT (557) and FE3DGW (558), can treat flow along all three dimensions of an aquifer. These models have been verified by comparison with data describing actual hydrologic systems (559-561).

The transport of contaminants, in particular the dissolved or entrained radionuclides, by a fluid flowing through a porous medium has had similar, but more recent, development. Like the analyses of fluid flow, analyses of contaminant transport are now part of the standard literature and the repertoire of students of water movement (562, 563). The available models range from one-dimensional codes, such as GETOUT, to multidimensional codes with convection, dispersion, and sorption properties that vary with position and time, such as MMT, SWIFT, FECTRA, and CHAINT (557, 564-567). Two codes, SWIFT and a recent extension of FE3DGW called CFEST, treat coupled hydraulic flow, contaminant transport, and heat transport.

Table II-11. Characteristics of Fluid-Flow and Contaminant-Transport Computer Models

Parameter	Fluid Flow			Coupled Flow and Transport		Contaminant Transport[a]			
	PATHS	VTT	FE3DGW	CFEST[b]	SWIFT	GETOUT[c]	MMT	FECTRA	CHAINT
Dimensions in space	Multiple	Multiple	Multiple	Multiple	Multiple	Single	Multiple	Multiple	Multiple
Flow properties	Constant	Variable	Variable	Variable	Variable	Constant	Variable	Variable	Variable
Dispersion				Variable	Variable	Constant	Variable	Variable	Variable
Radioactive decay[d]				Chain	Chain	Chain	Chain	Simple	Chain
Sorption				Variable	Variable	Constant	Variable	Variable	Variable
Release rate				Variable	Variable	Constant	Variable	Variable	Variable

[a]Values for the parameters describing fluid flow in these models must be supplied by a fluid-flow model.

[b]CFEST is operational but not yet documented.

[c]GETOUT theory has been extended to model variable flow, dispersion, sorption, and release rate, but the computer code is not yet operational.

[d]"Chain" means that the code accounts for nuclides produced during radioactive decay. "Simple" means that the code neglects such nuclides.

Sources: The information on the models shown in this table was drawn from multiple sources. The accompanying text discussion contains specific references for the models and topics addressed.

As contaminants move with ground water, they may be sorbed and held by the rock they are passing through. The parts of the contaminant-transport models that describe sorption assume equilibrium between the concentration of contaminants in the fluid and the concentration on the rock. The sorption parameter in these models is an empirical coefficient, called K_d, that is independent of the details of the actual mechanism by which sorption occurs. Improved models for sorption are currently being investigated as part of the NWTS Waste-Rock Interaction Technology Program (568). Short-term field verifications (569) of the contaminant-transport models have been performed. Additional verification of contaminant-transport models is planned as part of the NWTS Field Testing Program (570), described in Section II.F.2.

Because models of flow through fractures tend to be specific to particular types of fracture systems, they are less universally applicable than porous-media models. Although mathematical models of flow in fractured media are not so well developed as those for porous media and may not be needed for analyzing some repository sites, the amount of work done has been substantial (570-581).

There are two basic problems for the modeling of material transport in fractured media. One problem is the assembly of sufficient data to be able to adequately describe the actual hydrology of the far-field region around a repository site. The determination of effective permeabilities and fracture connections, although difficult, is possible. The second problem is the proper characterization of the sorption process in the fractured zones: although absorption may be dominant in porous rocks, fluid flow may be dominant in fractured rocks. Experimental work is being done on this problem (582), and the results will be included in future modeling. Until this work has been completed, the models for porous media are being used with equivalent formulations and conservative assumptions to predict bounds to the effects of flow through fractured media.

The field testing of transport for nuclides that travel very slowly requires such long times that few such tests are practical for model verification. Confidence in the results of the models may be gained, however, from comparisons with the observed transport of such nuclides from certain mineral deposits (583) and natural nuclear reactors (584, 585).

The output from the geosphere-transport codes is a prediction, as a function of time, of the radionuclide concentrations reaching the biosphere. As shown in Figure II-25, the two remaining steps in the performance assessment are calculations of radionuclide movement through the biosphere and of the radiation doses that people might receive. Available models perform both of these calculations. The doses from the ingestion of contaminated water and external exposures are calculated from codes such as ARRRG (586). Doses from contaminated foods are calculated from codes such as FOOD (587). The LADTAP code (588) calculates doses from the ingestion of water and fish and from external exposures. The doses from inhalation and from external exposures to acute and chronic releases into the air are calculated by codes such as DACRIN, KRONIC, and SUBDOSA (589-591). The predicted doses may be converted into predicted health effects (592). One aspect of biosphere-transport dose modeling that will be considered further as part of the Department's Waste Isolation Performance Assessment Program is the accumulation and dispersion processes associated with the chronic release of radionuclides into the biosphere over long time periods (593). The methods used in the biosphere-transport dose codes have gone through extensive development and verification over the past two decades and have been accepted in other NRC proceedings (588).

II.F.1.3 Methods for Assessing Near-Field Performance

Models for assessing repository performance in the near-field region must take into account mechanical stresses, heat flow, chemical interactions, and radiation. All of these phenomena, in addition to the properties of the host rock, affect the environment of the emplaced spent fuel. Chemical interactions at elevated temperature in the presence of radiation have already been studied in radiation chemistry. Although the stresses and the chemically active medium complicate the system, the expected interactions are understood by radiation chemists. Some specific physical interactions, however, have not yet been studied completely. For example, the thermomechanical effects of creep in salt and fracture in granite are understood only in part, although constitutive laws for these phenomena have been used in mine design. (Constitutive laws, or relations, express the dependence of strain on stress or, in other words, the way a material deforms under stress.) To improve the understanding of these effects, constitutive laws based on the laboratory study of rock samples under stress and temperature have been developed for salt (594); similar studies are being conducted to develop appropriate laws for granite, shale, and basalt.

The following sections discuss the status of the three principal types of models required for near-field analysis: heat-transfer models, thermomechanical models, and models of physical and chemical interactions among the emplaced waste, the components of the waste package, and the host rock.

II.F.1.3.1 Models for Heat-Transfer Analyses

Heat-transfer models are used to determine heat flow from the waste through the repository system. Analyses of heat flow are based on established theory (595) and are performed with verified computer codes.

A large number of thermal models have been developed through the years for various purposes. Many of these models have been coded for computers. Over 40 of them, in various stages of development, have been identified as useful in studies of waste disposal (596). Many of these 40 models also deal with other phenomena besides heat transfer. Available models can calculate temperature distributions and heat flow for almost all the configurations that may be used in waste repositories.

Many of the models, although developed independently, have overlapping or similar calculational capabilities. A comparison of the predictions made by various models can therefore give insight into the credibility of the predictions. Claiborne et al. (597) have compared calculations for a hypothetical repository using the codes entitled HEATING5 (598), THAC-SIP-3D (599), ADINAT (600), SINDA (601), TRUMP (602), and TRANCO (SPECTROM-41) (603). The agreement among the thermal models was generally good. In areas where the agreement was less than desirable, additional work is being performed to improve it.

Thermal models have also been used to predict temperatures in field tests of fuel assemblies in salt (116) and of heaters in salt (604), granite (582), and shale (605). Predictions based on generic thermal properties compared favorably with the experimental results. Predictions based on thermal properties determined from core samples or in situ probes agreed even better with the test results (606). An additional capability included in comprehensive thermal models like HEATING5 and TRUMP, is the prediction of the effect of ventilation in the repository (607, 608).

The temperatures of spent fuel rods within the waste package also can be calculated. Accurate determination requires the simultaneous calculation of convection, radiation, and conduction. The HYDRA-1 code (609) has been developed for this purpose. To verify this code, its predictions are being compared with experimental measurements for spent fuel currently stored in surface facilities. The magnitude of the convection coefficients that characterize heat transfer in a gas-filled waste package will be determined by experimental studies (610) currently being performed with scaled laboratory models (see II.F.2).

In summary, thermal models based on physical laws provide an accurate portrayal of heat flow and temperature. The experimental results obtained to date suggest that their precision can approach ±5%. This precision is important in ensuring that heat loads designed for the repository will not produce adverse effects in the host rock.

II.F.1.3.2 Models for Thermomechanical Analyses

The models developed for thermomechanical analysis are used to calculate the thermal and mechanical stresses and strain, on the repository system. Although these models can analyze stresses in engineered components, of principal interest in this section are predictions used to evaluate the integrity of the host rock.

Methods for the calculation of stress and strain in construction materials have been available for several decades. They are based on relations derived from physics and constitutive relations, or laws, obtained through laboratory tests. Because repository rocks are inhomogeneous and may be fractured, generic constitutive laws are more difficult to obtain for them than for most other construction materials.

The constitutive relations between stress and displacement are functions of temperature, load, load rate, strain rate, load path, and duration. For continuous media, constitutive laws can be formulated from continuum mechanics with the help of results from laboratory testing. For fractured media, there will be still more reliance on laboratory tests of material from specific sites, as well as on in situ tests, because it is difficult to derive accurate generic relationships. The inhomogeneity of rocks requires that these tests be conducted on larger samples than are normally used for engineering materials.

Such rock-mechanics testing will provide numerical values for the parameters in computer models. Field measurements from in situ tests will also help in verifying the models.

In their applications to repository design and performance assessment, thermomechanical codes are presently being used to analyze the following characteristics:

1. Uplift and subsidence.
2. Room closure and stability.
3. Hole closure and stability.
4. Canister movement.
5. Pillar stability.
6. Thermomechanical impact on ground water flow.
7. Stresses and strains at critical locations in the rock mass.
8. Failure of the rock mass.

A large number of models are available for using rock mechanics to predict the phenomena listed above (611); Table II-12 gives the current status of some of the more commonly used models. Among the characteristics listed in the table are the types of rock-deformation processes that the constitutive laws in each code can describe. Models of thermoelasticity depict rock deformations that are reversible: after an incremental force is removed, the rock will return to its original configuration. Thermal-creep and thermoplastic models depict time-dependent and time-independent irreversible deformations, respectively: after being deformed, the rock remains deformed. The ability to simulate these two types of deformation is an important property of the models because it affects their ranges of applicability.

The models vary in the extent to which they have been verified. The codes listed in Table II-12 have already been tested to some degree, although no single code can be considered to be verified for all aspects of the calculations it can perform.

As illustrations of the capabilities of these codes and of the ways they have been verified, the paragraphs that follow discuss three of the codes with references to more detailed discussions.

The TRANCO model (612), a thermal code, and the TEVCO model (613), a mechanical code, have been combined to form the SPECTROM series of codes, for which documentation is not yet available. The verification of the SPECTROM codes has been carried out by successful comparison of their results with analytical approximations (614) for the thermoelastic behavior of a repository. Results of the SPECTROM codes for a variety of problems also agreed well with results of the MARC-CDC codes, which are widely accepted for engineering calculations (615).

Further verification of the SPECTROM codes has come from comparing their predictions with field data from Project Salt Vault (616). A detailed laboratory analysis (594, 617) of the bedded salt in which that project was conducted provided input parameters for the code, which could then simulate the behavior of the bedded salt. Within the limitations of the assumptions made in modeling, the results were good, although the predictions of horizontal-pillar displacement and permanent deformation were found to be inadequate (616).

Table II-12. Characteristics of Some Commonly Used Rock-Mechanics Computer Codes

Code	No. of Dimensions in Space	Constitutive Relations[a]	Rock Types Analyzed	Coupling with Hydraulic Models
SPECTROM (a series of codes)	2 and 3	TE, TC, TP	Salt, sediments, granite, basalt, tuff	Under development
SALT 2, 3, 4 SALTY, REPOS EXPAREA	2 and 3	TE, TC	Salt, basalt,	Under development
STEALTH	2 and 3	TE, TC, TP	Salt, basalt, sediments, granite	Under development
SAP-IV PORFRC	2 and 3	TE, TP	Granite	Under development
SANDIA-BMINES	2 and 3	TE	Salt	None
COUPLEFLO	2	TC	Salt	None
SANCHO	2	TE, TC, TP	Salt and associated evaporites, tuff	None
FINEL BASFEH DAMSEL	2 and 3	TE, TP	Basalt	Under development
BAMBIT BEH2D	2 and 3	TE, TC, TP	Basalt	Under development

[a]The entries in this column are the phenomena that the constitutive relations describe. The abbreviations are TE for thermoelasticity, TC for thermal creep, and TP for thermoplasticity; these terms are defined in the text.

Sources: The information on the codes shown in this table was drawn from multiple sources. The accompanying text discussion contains specific references for the codes and topics addressed.

The SPECTROM codes have been used to predict the hole closure that might affect the retrievability of emplaced waste (618) and the displacements that occur in the room-and-pillar configurations of a salt repository (619). The codes also have studied the effects of varying the parameters in the analysis of the room-and-pillar configurations (460); such parametric studies provided insight into the accuracy required of model parameters.

The STEALTH code (620), used primarily for the analysis of salt repositories, has been used to investigate the creep phenomenon associated with salt (621, 622). To verify this code, a three-dimensional simulation of Project Salt Vault has been carried out (623), using constitutive laws that have been compared with laboratory data. The results generally agreed well with the field data. Further verification is under way.

Another aspect of rock mechanics that is now being addressed is the thermomechanical impact on ground water, especially in fractured media. Basic field data on the hydraulic properties of fractures (254) and the fractures observed in heater experiments have been gathered at the Swedish Stripa mine (624). Preliminary calculations have been performed and evaluated for a simplified repository in fractured rocks (625). The STEALTH code is being used to couple thermomechanical response with ground water flow (626).

As explained in more detail in Section II.F.2, laboratory tests, bench-scale tests, and field tests now in progress are producing data that agree well with predictions of thermomechanical models. Enough experience in the verification of rock-mechanics codes has been gained to permit an estimate of the time required for resolving the remaining technical questions listed in the Earth Science Technical Plan (627) prepared by the Department and the U.S. Geological Survey. Experience also indicates that the models of thermomechanical impacts on ground water will be fully verified by the end of FY 1987 (628). The near-field thermomechanical models will be accurate enough to design repository systems with adequate margins of safety.

II.F.1.3.3 Models of Interactions Among Package Components and Rock

To predict the near-field behavior of a repository requires analyses of the interactions among the emplaced package components and the host rock. These interactions fall into six general classes of problems: (i) the movement of fluids like ground water to the vicinity of the waste package, (ii) the corrosion of the canister and sleeve materials by these fluids or by the local fluids in the rock, (iii) the dissolution of the waste matrix by ground water containing the added corrosion products, (iv) the sorption of radionuclides in the rocks and in the engineered systems, (v) the absorption of radiation emitted from the waste, and (vi) the alteration of chemical phases and properties in the vicinity of the canisters. A study of these interactions predicts the kinds, amounts, and chemical state of the radionuclides available for entry into the ground water system.

Investigations of the mobilization of fluids in the vicinity of the waste package have supported the development of models, both theoretical and experimental, of the transport of these fluids to the waste package (see II.F.2) (608, 629, 630). A computer code, entitled MIGRAIN (608), is available for predicting the transport of brine through salt. Many of the far-field models of fluid flow (see II.F.1.2) are useful in modeling the near-field flow of ground water. In addition, the codes GWTHERM (631, 632) and PORFRAC (633) are particularly useful in modeling near-field fluid flow at elevated temperatures or in rock fractures.

Each of the other five problems can be separated from the others well enough for individual, but not isolated, study.

The study of corrosion allows modeling of the metallurgical behavior of the canister materials in intrusive ground waters (634). Dissolution studies, using fluids modified by the corrosion processes, allow modeling of the leaching process by which fluids take up radionuclides from the waste (635). Sorption studies, using the fluids modified by the corrosion and dissolution processes, permit modeling of radionuclide sorption (636-640) in backfill materials and the host rock. The sorption process is included in the hydrologic-transport codes discussed in Subsection II.F.1.2. Studies of the absorption of radiation proceed from a description of the radiation emitted by the spent fuel, which the ORIGEN code (641) calculates. Codes like ANISN (642) and DOT (643) can then calculate the energy deposited in the materials around the waste.

The data that support the models used in these studies are leaching and dissolution data for nuclide release; canister, filler, overpack, and sleeve behavior for the definition of corrosion; data on ion exchange, precipitation, and chemical and physical sorption for the sorption studies; and data describing the chemical phases of the material at the appropriate temperatures and pressures for the definition of the physicochemical state.

The current status of mathematical models describing the chemical interactions that occur in geologic environments was recently reviewed (644). Most of the models assume equilibrium conditions and rely heavily on available data to describe the following reactions: dissolution, precipitation, the formation of soluble complexes, and the formation of soluble free ions. A few models also describe such reactions as ion exchange or the sorption of a solution species onto a solid adsorbent. Most of the models can address temperature variations if the necessary input data can be supplied as a function of temperature. A few models can address the time dependence (kinetics) of chemical reactions, but since few data are available on the kinetics of important geochemical reactions, little effort has been placed in its computation.

There is a continuing effort to adapt existing chemical models for use in the NWTS Program. In the past few years, the three chemical models discussed in the paragraphs that follow have been studied to ascertain their usefulness for analyzing the chemical reactions that may occur in the near field.

Early attempts to show the applicability of chemical thermodynamic modeling to processes under deep-repository conditions used a computer model called FASTPATH (645, 646). The preliminary work with FASTPATH attempted to simulate the chemical reactions expected between ground water and the basalt in the Pasco basin in the State of Washington (647, 648), with and without the presence of carbon dioxide at temperatures ranging between 25^{o}C and 300^{o}C. Predictions made with the model were compared with observed reactions between ground waters and minerals from deep borings on the Hanford Site. Comparisons, although not exact, were favorable enough to conclude that further effort was warranted.

A similar thermodynamic model, EQ3/EQ6 (649, 650), has some potential advantages over FASTPATH; it will be able to address alteration reactions and the weathering of rock by water and alteration reactions. This model can be adapted to include sorption reactions as well as kinetics if the appropriate data become available. A third model, MINEQL (651), is being studied to assess whether it can successfully predict nuclide sorption.

It is planned that by the end of FY 1982 the capabilities of these three models will have been determined. They then may become subroutines within larger computer codes describing hydrologic mass transport. The coupling of chemical models with hydrologic mass-transport models has been demonstrated in the analysis of the shallow land burial of radioactive wastes at the Hanford Site (652, 653).

The data needed for describing the basic thermodynamic properties of some actinides, fission products, and minerals of crystalline rocks have been identified (654). Some of these data are being obtained in NWTS-sponsored programs (655, 656).

II.F.1.3.3 Development of More Versatile Near-Field Models

Current work in refining and coupling the individual models is in progress to develop more complete, more versatile models describing complex phenomena that are not independent. These advanced models will be available in 1981, and the coupling will be done by 1982. Insights gained from sensitivity and uncertainty analyses will be used to plan further development, and verification in laboratory and field tests will continue throughout the process. A group of scientists from the Department and the U.S. Geological Survey has concluded that coupled and verified models describing interactions between waste and rock will be available in 1985 (657).

II.F.1.3.4 Conclusions

The development and verification of individual models of near-field phenomena are sufficiently advanced for their application to analyses of repositories. At their current state of development, the models can show the major effects that can occur in the near field. Their results can be used in generating conservative inputs to analyses that predict the overall long-term performance of repository systems. Further refinements and coupling of the codes, which the Department's program emphasizes, will improve the details of their predictions and will make possible less conservative, more realistic analyses of overall long-term performance.

II.F.1.4 Applications of the Performance-Assessment Methods

This subsection begins by summarizing the reported applications of the performance assessment methods and by drawing two conclusions from them. The subsection then presents summaries of four studies that use the methods.

The first of the examples is from an analysis conducted as part of the program of the International Nuclear Fuel Cycle Evaluation; it examines a scenario describing the transport of radionuclides by ground water from a hard-rock repository. The near-field conditions assumed in this study are appropriate to a waste disposal system that emphasizes engineered barriers against radionuclide transport. The second example is from a project being conducted within the NWTS Program; it examines a similar scenario for a repository in a salt dome. The conditions assumed in both analyses were chosen as typical of the conditions that could be encountered at a potential repository site, but there was no attempt to choose conditions characteristic of sites of high quality. For example, the conditions for the salt dome site were chosen because the information available in the literature for an actual dome was sufficient to allow the analyses to be performed with a minimum number of assumptions about unavailable data. The analyzed dome does not meet the NWTS siting criteria in Section II.D.3 and is not under consideration as a potential site. The third example is a state-of-the-art application of transport modeling to an existing problem in radionuclide contamination. The fourth example is from a project that is evaluating the performance of waste packages in a geologic repository.

II.F.1.4.1 Conclusions Drawn from Reported Applications

A number of analyses of performance assessment have been reported (382, 637, 658-669). These studies range from simple analyses using only a few mathematical relations to complex analyses using the models described in the preceding sections of this chapter.

As the exploration and characterization of sites continue under the NWTS Program, many more studies will be performed to help in the major tasks and decisions described in Subsection II.F.1.1.3. The existing studies show, however, that the methods of analysis are ready for application to actual and hypothetical waste disposal systems, even though the models will be improved and made more complete in the next several years.

The existing studies also support a second conclusion. They show that the work done so far on waste disposal systems has identified no insurmountable difficulties in ensuring that the systems will be safe. The studies have examined the consequences of many phenomena that might release radionuclides from a repository. Although these phenomena are unlikely at sites that have been chosen in accordance with the criteria in Section II.D.3, the studies have recognized transport by ground water as the principal naturally occurring phenomenon by which radionuclides might leave a repository. For this reason, several studies have examined water intrusion and the subsequent transport of radionuclides. Some of these studies have been specific to actual or proposed sites; other studies have been generic, intended to predict in general terms the performance of a broad class of repositories.

The studies of ground water transport predicted the radiation doses that might be delivered to people in the assumed scenarios. The predicted doses would meet existing radiation-protection standards for other fuel-cycle facilities and would meet Objective 2 (II.A.1); that is, the doses would generally be lower than the doses delivered by natural background radiation. For example, an analysis of a small spent fuel repository in bedded salt at a particular site in New Mexico has shown that even an incredibly large flow of water through the repository would deliver doses that would be less than 3% of those delivered by natural background radiation (670).

Another study, describing radionuclide releases by ground water flow through a salt dome, is summarized in II.F.1.4.3, below; that study also predicts doses much below those delivered by natural background radiation. Although more studies remain to be done, analyses to date have not predicted large radiation doses to people after ground water intrudes into a repository.

It is possible to predict much higher doses by assuming events that are even more unlikely than the intrusion of ground water. Such doses have been reported in studies that examine the worst consequences of repository breaches without regard to the credibility of the breaches. For example, the impact of a giant meteorite on a repository might deliver doses approximately 1,000 times higher than those from natural background (659). Such a scenario for radionuclide release is not credible, however. It assumes a highly improbable event that might be expected to occur at a repository site less than once in 1 trillion years; furthermore, it assumes that people live around the site during and after the cataclysmic event.

Although most future actions by people would not breach a repository seriously enough to deliver large radiation doses to people, a few studies of such actions have been able to predict large doses by assuming conditions that seem to be unrealistic or incredible. The most striking example of these studies examines the consequences of future solution mining in a salt dome containing a sealed and forgotten repository (668). Under the assumptions made in the study, the predicted doses are clearly unacceptable. The study predicts these large doses, however, by assuming that no measures such as markers or records, would be taken to warn future societies (see II.E.3); that no engineered barriers, with the containment characteristics found achievable in Section II.E.1, would be present; and that the solution mining would continue for 50 years without the discovery that the repository was present. The scenario modeled in this study was developed as part of a larger effort to determine the potential importance of human intrusion and to determine how an unprotected system might perform. The objective of the larger effort was to determine appropriate protective measures.

Although unequivocal statements about the safety of mined repositories will be possible only for specific sites, the performance assessments of reference systems indicate that properly chosen and designed systems would meet the objectives stated in Section II.A.1.

II.F.1.4.2 Example 1: Analysis of a Hard-Rock Repository

The International Nuclear Fuel Cycle Evaluation study has evaluated the health and safety impacts of seven different fuel cycles (666), one using light-water reactors with the disposal of spent fuel. The analysis for this cycle is summarized here; important features are abstracted from the full report (667).

In this analysis, all radioactive wastes from the cycle, except for mining and milling wastes, were assumed to be placed in a geologic repository in hard crystalline rock, either granite or gneiss. These rocks are in many respects representative of a broad class of hard, crystalline silicate rocks (671).

The site of the reference repository is composed of large areas of granite or gneiss consisting of solid rock blocks surrounded by small fracture planes or joints that are interconnected. It is generally accepted that the hydraulic conductivity of these large areas, where flow is through joint systems, is less than 10^{-11} m/sec and that, therefore, significant flow can occur only through the surrounding highly fractured zones (672).

The extent of the area used for the far-field modeling is 25,000 by 25,000 m. Lakes and rivers are included within this area. The mathematical model of the area includes a description of the topography and structural properties. The ground surface elevations represent the ground water elevations throughout the region; a ground water divide to the south has an elevation of 40 to 45 m above the level of the sea to the north. The major fracture zones are represented by discrete elements.

The underground part of the repository consists of a tunnel system in the hard crystalline rock, 500 m below the surface. Its dimensions are based on the ability to hold the waste arising from 1 year of a 100-GWe economy; the dimensions are 1,150 by 1,150 by 12.4 m (673). All categories of waste are disposed of in the repository, although the various types of waste are placed in separate parts of the repository. In placing the types of waste, due regard is given to the direction of ground water flow; in order to avoid adverse geochemical reactions, leachate from one type of waste is prevented from reaching other types.

It is assumed that spent fuel is placed in stainless steel canisters filled with lead and positioned in the repository. The areas around the canisters and the tunnel are backfilled, respectively, with pure bentonite and a mixture of bentonite and sand (674). The medium-level and low-level wastes are mixed with concrete and packed in drums for placement in the repository.

The presence of a repository will perturb the host rock, but the repository design attempts to minimize adverse impacts. The removal of rock for waste emplacement creates new hydrologic paths and openings that will not close because of the low extent of plastic deformation; adverse effects are minimized by the proper orientation of tunnels and by backfilling the tunnels and openings. The compacted bentonite and the bentonite-sand mixture swell upon absorption of natural ground waters, and their permeability decreases.

Heat-induced effects are minimized by proper heat loading, achieved by basing the spacing of the waste canisters on the conductivity and heat capacity of the rocks. The spent-fuel canisters are placed singly in vertical holes 3.5 m apart in tunnels that are 25 m apart (675).

II.F.1.4.2.1 Scenario Development

The data on hard crystalline rock indicate that, except for highly improbable events like meteorite strikes and volcanic eruptions, no geologic activity like tectonics or erosion could be expected to disrupt the repository. If appropriate site-selection criteria and protective measures are applied (II.E.3), the probability of deliberate or inadvertent human intrusion into a repository or its immediate environs should be negligible. For these reasons, man-caused disruptive scenarios were not considered in this study (676). As a result, the only release studied arises from a scenario in which wastes are transported by the small amounts of water normally present in crystalline rocks at depth. The transport occurs after the waste canisters have failed.

In the reference repository, released radionuclides move from the waste through the compacted bentonite backfill and into ground water within rock fissures. Because canisters are placed in individual holes, the release rate is estimated by taking into account the resistance to mass transfer of the bentonite buffer. It is assumed that the bentonite is homogeneous and that the canister hole intercepts a set of parallel horizontal fissures in which ground water flows around the bentonite annulus. Because of the low permeability of compacted bentonite, it is assumed that water flow in the annulus is too small to increase the mass transfer significantly beyond that due to diffusion alone. No ion-exchange capacity is assumed for the bentonite to allow for the potential inhibiting influence of corrosion products. The conditions required by Objective 1 in Section II.A.1, regarding the containment of waste, were not assumed. The canister lifetime is assumed to be 100 years in this example, and simultaneous failures of the canisters take place at 100 years.

Because spent fuel is predominantly uranium oxide, it was assumed that dissolution occurs at a rate corresponding to the solubility-limited diffusion of uranium (677). Other radionuclides are released in proportion to the ratio of their concentrations to the concentration of uranium. Because some iodine tends to migrate out of the fuel pellets during burnup, it is assumed that 10% of the inventory of iodine-129 is initially not bound to the uranium matrix and is immediately mobile inside the bentonite backfill.

II.F.1.4.2.2 Methods of Analysis

The prediction of radionuclide transport in this study requires an estimate of water movement because water is the transport medium. As described in Subsection II.F.1.2, hydraulic models predict ground water flow paths and travel times from input data describing the system and the release scenario. In this example only normal slow water movement through the very impermeable rock is considered. The three-dimensional flow model FE3DGW is used (678) to analyze the water movement; it is able to treat the effects of highly fractured zones on regional water flow and the decrease in permeability and porosity of the rock masses and fracture zones. Output from the flow model is used as input to the one-dimensional transport model GETOUT to analyze radioactive decay and interactions among rocks, nuclides, and water. Best-estimate values of distribution coefficients based on laboratory data (679) are used for determining the retardation of the nuclides. The output from the transport model is nuclide concentrations in the ground water as a function of time.

Radionuclides in the ground water are assumed to enter the biosphere as seepage into a fresh water lake. Subsequent transport in the biosphere is modeled by a multicompartment model with parameters selected to represent the reference granitic site. The model of the biosphere is divided into three subsystems of progressively increasing size representing regional, intermediate, and global ecosystems. The structure of this model permits recirculation of radionuclides between subsystems. The regional ecosystem includes the fresh water lake that receives the radionuclides, lake sediments, soil, and subsurface ground water. The intermediate ecosystem is a large lake or sea with associated sediments. The atmosphere above the regional intermediate areas is modeled to an altitude of 1 km. The global ecosystem models the oceans, the continents, and the global atmosphere.

Calculated concentrations of radionuclides in the biosphere are used in the exposure-pathway analysis to estimate the total intake by a maximally exposed person. The pathways for intake include inhalation, the ingestion of food and drinking water, and external exposure from material deposited on the ground. The pathway analysis provides the external dose and the inhalation and ingestion rate for each radionuclide. The intake rates are used to calculate weighted whole-body doses for each radionuclide.

II.F.1.4.2.3 Results

For the thermal loadings of 10-year-old spent fuel used in this example, the predicted peak temperature at the canister surface is about 80°C; maximum and mean temperatures in the host rock are 62°C and 52°C, respectively. These three temperatures occurred respectively at about 10, 25, and 90 years after emplacement in the repository (680).

For the leaching or release-rate analysis, the release was assumed to begin at 100 years. The calculated leach times were 200,000 and 230,000 years (681) for PWR and BWR spent fuel, respectively.

The hydrologic analyses predict an average path length of 7,100 m, an average travel time of 11,700 $\pm$1,300 years, and an average velocity of 0.61 m/yr for the reference hard-rock site (682).

The maximum nuclide discharge rates to the biosphere and their dependence on time are shown in Table II-13. The nuclide travel times are generally on the order of hundreds of millions of years, permitting most of the nuclides to decay. An exception is iodine-129, which migrates at the ground water velocity with a travel time of about 10,000 years.

Table II-13 also shows the maximum annual doses received by a person from the spent fuel and the associated waste. These doses are whole-body doses received by the maximally exposed person during the 50th year following continuous exposure for 50 years.

All discharge rates to the biosphere calculated by the transport model are small. The earliest peak concentration observed is that for iodine-129 at 11,000 years for release from the spent fuel. No other significant fission product peaks appear. The peak concentrations for the actinides all occur at 400 million years or later. Because of long transport times, the significant discharge rates to the biosphere arise from the long-lived nuclides uranium-235 and uranium-238 and their daughter nuclides.

The major contribution to dose comes from the radionuclide protactinium-231, with minor contributions from radium-226. All doses are small fractions of the average annual background radiation dose (0.1 rem) and well within the requirements of Objective 2 (II.A.1).

Table II-13. Maximum Discharge Rates, Times of Maximum Discharge, and Dose Rates Received by People from a Hypothetical Spent-Fuel Repository

Radionuclide	Time of Maximum Discharge (years)	Maximum Discharge Rate (Ci/yr)	Maximum Dose Rate from Spent Fuel (rem/yr)	Maximum Dose Rate from Associated Waste (rem/yr)
^{129}I (a)	1.1×10^{4}	5.6×10^{-4}	-	-
^{129}I (b)	1.1×10^{4}	1.3×10^{-2}	5.1×10^{-6}	-
^{135}Cs	7.1×10^{8}	6.3×10^{-11}	1.5×10^{-14}	-
^{226}Ra (c)	4.1×10^{8}	9.4×10^{-5}	1.0×10^{-7}	1.6×10^{-6}
^{230}Th	4.1×10^{8}	1.0×10^{-4}	2.8×10^{-8}	4.5×10^{-7}
^{231}Pa	4.1×10^{8}	2.2×10^{-5}	5.7×10^{-5}	5.5×10^{-5}
^{232}Th	8.1×10^{8}	3.5×10^{-8}	4.7×10^{-10}	-
^{234}U	4.1×10^{8}	2.0×10^{-4}	1.4×10^{-8}	7.0×10^{-8}
^{235}U	4.1×10^{8}	1.0×10^{-5}	3.0×10^{-9}	3.0×10^{-9}
^{236}Th	4.1×10^{8}	2.0×10^{-9}	2.7×10^{-13}	-
^{238}U	4.1×10^{8}	2.0×10^{-4}	2.4×10^{-8}	1.2×10^{-7}
^{230}Th/^{226}Ra (d)	4.1×10^{8}	-	2.2×10^{-6}	1.1×10^{-5}
^{234}U/^{226}Ra(e)	4.1×10^{8}	-	5.5×10^{-8}	2.7×10^{-7}
Maximum annual total dose	4.1×10^{8}		5.9×10^{-5}	6.9×10^{-5}

[a] Bound in fuel matrix.
[b] In gap between fuel cladding and matrix.
[c] Radium-226 reaching the biosphere directly from ground water.
[d] Radium-226 produced by the decay of ^{230}Th in the biosphere.
[e] Radium-226 produced by the decay of ^{234}U (via ^{230}Th) in the biosphere.

Sources: (Reference 683) Working Group 7, International Nuclear Fuel Cycle Evaluation Committee, Release Consequence Analysis for a Hypothetical Geologic Radioactive Waste Repository in Hard Rock, INFE/DEP/WG7/21, Table XIX, December 1979

(Reference 684) Working Group 7, International Nuclear Fuel Cycle Evaluation Committee, Release Consequence Analysis for a Hypothetical Geologic Radioactive Waste Repository in Hard Rock, INFE/DEP/WG7/21, Table XXV, December 1979

II.F.1.4.3 Example 2: Analysis of a Salt-Dome Repository

II.F.1.4.3.1 Introduction

This analysis examines one of several scenarios reported in a study (668) performed for the NWTS Program by the Pacific Northwest Laboratory and the Office of Nuclear Waste Isolation as part of an effort to integrate safety-analysis techniques for waste repositories. The data used in the analyses are for the most part from an existing salt dome that has been disqualified from consideration as a repository site.

The repository is approximately 640 m below ground level in the salt dome, the top of which is 400 m below ground level. At the repository horizon the dome is approximately 3,400 m in diameter. The area excavated for the repository is separated from the edge of the dome by a 240-m buffer zone. There are two aquifer systems, which are hydraulically separated by an aquitard. The upper aquifer is fresh water and is under a greater hydraulic head than the lower fresh water aquifer. The upper aquifer is approximately 270 m above the top of the dome. The salt dome intersects the lower aquifer, which extends around and above the dome.

The underground part of the repository consists of a tunnel system on one level 640 m below the surface. Spent fuel is placed in vertical holes in tunnel floors on 1.67-m centers over the 1,375-acre repository area to distribute the decay heat load (685). A total inventory of 90,000 MT of spent fuel is assumed to be emplaced in the repository. No high-level waste other than spent fuel is disposed of in the reference repository (686).

The spent fuel is encapsulated in 1.27-cm-thick carbon-steel canisters filled with air and spaced in the repository to obtain a thermal loading of 60 kW/acre. The backfill around the canisters and in drifts consists of crushed salt (687). No credit is taken in the analysis for the engineered systems required by Objective 1 or for the protective measures against human intrusion described in Section II.E.3.

II.F.1.4.3.2 Scenario Description

Only a human breach of this isolation system is postulated in the analysis because no credible natural events were identified as capable of breaching the repository within 1 million years. The reference dome is assumed to be inadvertently breached by exploratory drilling; no corrective action is taken. The abandoned borehole connects the two fresh water aquifers. Because no credible failure mechanism could be identified for connecting the repository to the lower aquifer 100 years after repository closure, the study simply postulates a breach through the buffer zone to the aquifer. Thus, hydraulic continuity is assumed to be established through the repository system (688).

It is assumed that the spent fuel has decayed for 100 years at the initiation of the scenario. The dissolution rate of the spent fuel is calculated to be controlled by the solubility limit of 6 ppm for uranium in ground water (689). Other isotopes are assumed to be carried into the solution in proportion to the amount of uranium that was dissolved (690).

The head differential between the two aquifers is approximately 50 ft (15.2m), with flow from the upper to the lower aquifer (691). It is assumed that the flow continued for a period of about 15,000 years, at the end of which time fresh water flow has been sufficient to dissolve the entire dome immediately above the repository emplacement area (692). The resultant dome collapse over the emplacement area results in decreased fresh water flow because the flow must pass through subsurface rubble (693). Contaminated water from the dome is calculated to flow with the ground water through the lower aquifer to a river 100 km (62 miles) from the dome (694).

II.F.1.4.3.3 Method of Analysis

The method of analysis for radionuclide transport from the salt dome repository is similar to that for the hard rock example in that it requires knowledge of both near-field and far-field hydrology, radionuclide decay and interactions with the natural materials, and pathways to people. To determine the flow through the dome, the FE3DGW code is used (695).

The calculated flow rate and field data around the reference dome are used in the VTT computer code (696) to estimate regional ground water movement and in the MMT computer code (697) to determine transport times. The transport calculation was done in one dimension. Values of the retardation coefficients used in the MMT model are based on aquifer data representative of the region of interest, taking into account salinity increases induced by the scenario (698).

Radionuclides from the ground water enter the biosphere as seepage into the river 100 km away. Maximum individual doses and population doses for persons living along the river are determined using a modified version of the LADTAP (699) code, taking into account the consequences of eating fish, drinking water, and external exposure.

Assumptions relevant to the scenario and the calculations are made conservatively; they produce overestimates of the radionuclide releases from the system.

II.F.1.4.3.4 <u>Results</u>

For the 50-ft (15.2-m) differential head, the FE3DGW code predicts a rate of flow through the dome breach of approximately 260 gpm (68.7 l/m). Following the collapse of the upper dome at 15,000 years, the flow rate is calculated to be reduced to 36 gpm (9.5 l/m) (688). The far-field analysis by the VVT code uses the path length of 326,000 ft (99,104 m) to the river, along with an average ground water velocity of 7.6 ft/yr (2.3 m/yr), to predict a ground water travel time of 43,000 years to the river (700).

The resulting maximum nuclide discharge rates and their travel times for the nuclides to the river, as predicted by the MMT code, are shown in Table II-15 (701). The travel times are based on retardation coefficients established by laboratory measurements (702). While those coefficients are only approximate, they have been chosen to provide a conservative assessment of radionuclide migration from the reference site. The doses received by the maximally exposed person (699) are also presented in Table II-14.

Table II-14. Maximum Discharges, Times of Maximum Discharge, and Maximum Dose Rates from a Hypothetical Spent-Fuel Repository

Radio-nuclide	Maximum Nuclide Concentration (microCi/ml)	Time That Max. Nuclide Concentration Occurs (years)	Maximum Individual[a,b] Whole-Body Dose Rate (mrem/yr)
^{14}C	3.4×10^{-8}	45,000	5.0×10^{-4}
^{99}Tc	7.7×10^{-4}	58,600	1.1×10^{-0}
^{129}I	2.0×10^{-7}	50,500	7.7×10^{-4}
^{135}Cs	2.7×10^{-6}	45,000	4.8×10^{-2}
^{236}U	9.6×10^{-7}	440,000	5.4×10^{-2}
^{232}Th	1.2×10^{-13}	2,000,000	1.1×10^{-8}
^{237}Np	7.4×10^{-6}	44,000	1.9×10^{-1}
^{233}U	1.9×10^{-7}	1,690,000	1.2×10^{-2}
^{229}Th	6.8×10^{-9}	1,500,000	1.4×10^{-3}
^{238}U	8.2×10^{-7}	438,000	4.1×10^{-2}
^{234}U	2.0×10^{-6}	437,000	1.2×10^{-1}
^{230}Th	2.1×10^{-8}	520,000	6.3×10^{-4}
^{226}Ra	6.3×10^{-8}	490,000	$1.5 \times 10^{+1}$
^{243}Am	1.7×10^{-6}	45,000	4.5×10^{-2}
^{239}Pu	8.5×10^{-10}	80,000	6.8×10^{-6}
^{235}U	1.9×10^{-8}	470,000	1.0×10^{-3}
^{231}Pa	3.0×10^{-10}	460,000	1.3×10^{-1}

[a]The maximum individual dose rate generally occurs in the child age group.

[b]For some isotopes, the dose rate to individual organs may be higher.

Sources: (Reference 700) M.A. Harwell et al., Reference Site Initial Assessment for a Salt Dome Repository, PNL-2955, (Working Document), Appendixes G and H, Pacific Northwest Laboratory, Richland, WA, December 1979;

(Reference 700) Ibid., pp. 11-47 to 11-49;

(Reference 699) W. C. Arcieri, J.G. Feinstein, and J.D. Freeman, Methods Used to Compute the Dose to Individuals and the General Population Due to Geotransport of Radionuclides from a Spent-Fuel Repository, NUS-TM-327 (Draft Report), NUS Corporation, Bethesda, MD, February 1980.

Additional analyses have been performed to determine whether a change in the spent-fuel emplacement configuration accompanying immersion in a brine solution could lead to nuclear criticality as a result of salt dissolution. Criticality calculations have been performed with the NUMICE-2, LEOPARD, and PDQ-7 (703) computer codes. Evaluations were performed with and without engineered barriers, taking into account the salinity of the brine solution (704). The results of the analysis show that spent fuel with a burnup of 33,000 MWd/MT does not present a criticality problem even if the waste is dissolved in a pocket of brine (705).

In defining the reference scenario, several natural phenomena were considered. Such phenomena as climatic fluctuations and glaciation (706), denudation, erosion, and sedimentation (707) were eliminated as potential significant release mechanisms because of the planned depth of the repository (600 m). Because geologic siting criteria are applied to the selection of the repository site, the following potential release mechanisms were judged to have less serious potential consequences than the postulated mechanism: surface flooding (708), igneous activity (709), diapirism (710), diastrophism (711), diagenesis (712), metamorphism (709), fracturing (713), and fissuring, faulting, and seismicity (714).

Meteorite impact was eliminated as a potential problem because of the low probability of impact by a meteor of sufficient size to cause significant damage to the repository (713).

The abandoned borehole assumed in this study also represents other conditions, for instance, undetected openings and leaks around shaft or borehole seals. The abandoned, unplugged borehole, drilled after repository closure, was considered a worst-case approximation whose consequences would be expected to exceed those that could reasonably be expected to occur.

The analysis takes no credit for engineered barriers and protective measures against human intrusion. For this reason, the analysis is considered to produce an overestimate of releases and an overestimate of the credibility of their occurrence. Nevertheless, the doses predicted in the far-field analysis comply with Objective 2.

II.F.1.4.4 Example 3: Modeling of Hydrologic Transport

In contrast to the two preceding studies, which examine hypothetical repositories, the third example illustrates not only the use of models but also their verification. The third example involves monitoring and modeling the transport of very small quantities of radioactive and chemical waste in an aquifer. The waste is produced by the cleanup of process water from nuclear facilities at the Idaho National Engineering Laboratory (INEL) and discharged into disposal ponds and injection wells and from these points into the aquifer. These waste water streams are within standards both at the point of injection and at the points of reuse. The Department is examining alternatives, such as recycle and further treatment before discharge.

This example, taken from the published literature, is an application of hydrologic transport modeling to an actual problem: the transport of radioactive waste away from disposal ponds and a disposal well. The work was published in two parts (569, 669). The first part of the study models movement through the overlying strata to the aquifer; the second part models movement in the aquifer. The studies are of special interest because they involve both the transport of radioactive materials and the field verification of the results of the model.

The Laboratory occupies about 2,300 km^2 of semiarid land in southeast Idaho on the eastern Snake River Plain. This site is operated by the Department for the testing of nuclear reactors, for the fabrication and reprocessing of certain nuclear fuels, and for the temporary surface storage of certain defense wastes. Several of the facilities there produce and discharge low-level radioactive wastes to the subsurface by means of seepage ponds or wells. Two of these facilities, the Test Reactor Area and the Idaho Chemical Processing Plant, were considered in this study. The report describes the hydrologic system as follows (669):

> The eastern Snake River Plain is a large structural basin . . . that has been filled to its present level with perhaps 5,000 ft of thin basaltic lava flows and interbedded sediments. Nearly all of the

> eastern Snake River Plain is underlain by a vast ground water reservoir known as the Snake River Plain aquifer, which may contain in excess of 1 billion acre-feet of water. The flow of ground water in the aquifer is principally to the southeast at relatively high velocities (generally 5 to 25 ft/day) . . . eventually entering the Snake River Plain aquifer zone 450 ft below the ponds.

For the first study, a three-part numerical model was developed to simulate percolation through the interlayered basalt and sediments into an underlying lens of water. The model includes the effects of convection, dispersion, adsorption, and radioactive decay. The first part of the model simulates water flow and solute transport from the ponds to the lens. The second part simulates two-dimensional horizontal flow and transport in the lens, ensuring complete mixing. The third part of the model simulates vertical flow and transport from the lens toward the aquifer; this movement occurs through leakage to the underlying basalt (the Snake River Plain aquifer).

For the second study, which treats movement in the Snake River Plain aquifer, a hydrologic model was constructed for an area of approximately 6,600 km^2, less than 25% of the total aquifer area. The effects of areas of the aquifer outside the area of interest are minimized through adjustment of the boundary conditions. The model assumes that the flow satisfies Darcy's law, a fundamental mathematical relation in the description of fluid flow. The hydraulic part of the model was verified by comparing its predictions with the known location of the local water table. It was further verified by comparing its predictions of transient responses with the actual observed responses. The transport of solute was modeled only in the area of contamination.

The amounts of radionuclides discharged into the disposal ponds are known. When this information is used in the models, the distributions of transported material in the aquifer can be predicted. These distributions have been measured by actual field sampling. The hydraulic parts of the model were verified by calculating the distribution of chlorides and comparing it with the measured distribution; this comparison also yielded values for dispersivities, numerical values that characterize the transport. With these dispersivities, the movement of tritium was modeled. When sorption was included in the model, the study was able to simulate the movement of strontium-90. Calculations also were done for other nuclides, including cesium-137, but the cesium was so strongly sorbed as to be unavailable in the sampling much beyond the sediments in the discharge ponds.

Figures II-26 and II-27 shows the calculated and the measured distributions of tritium and of strontium-90. These results verify the model within the limits of its assumptions.

In addition to these studies at the Idaho National Engineering Laboratory, similar work has been performed elsewhere. The migration of a tritium plume has been studied at the Hanford Site (560). Auburn University has conducted an experiment on heat storage in an aquifer; the flow of the heated water was predicted by the contaminant-transport methods (561). The results of these studies also agree with predictions.

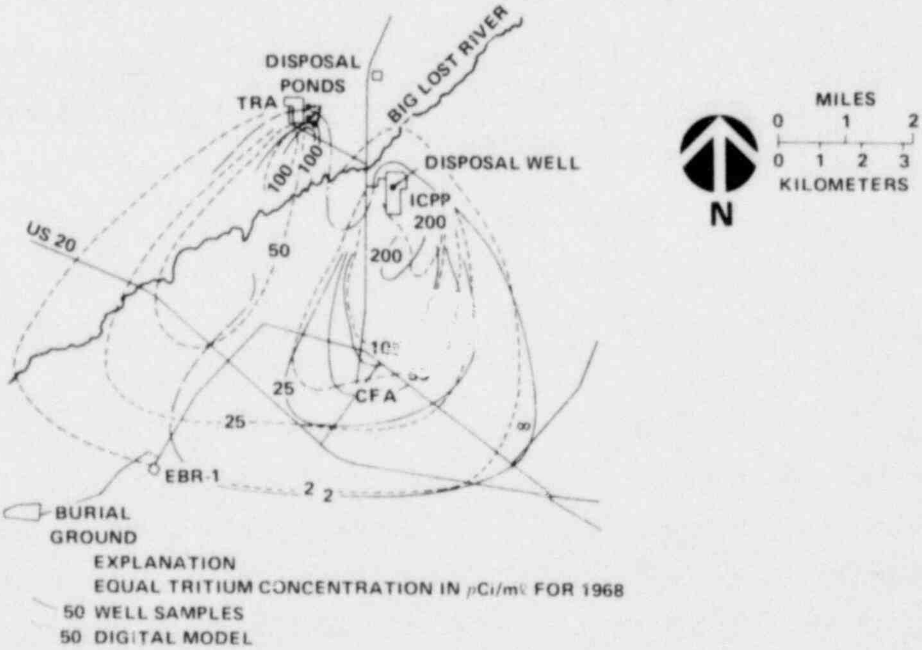


Figure II-26. Comparison of Computed and Measured Tritium Concentrations

Source: (Reference 669) J.B. Robertson, Digital Modeling of Radioactive and Chemical Waste Transport in the Snake River Plateau Aquifer at the National Reactor Testing Station, Idaho, IDO-22054, U.S. Geological Survey, Idaho Falls, ID, 1974

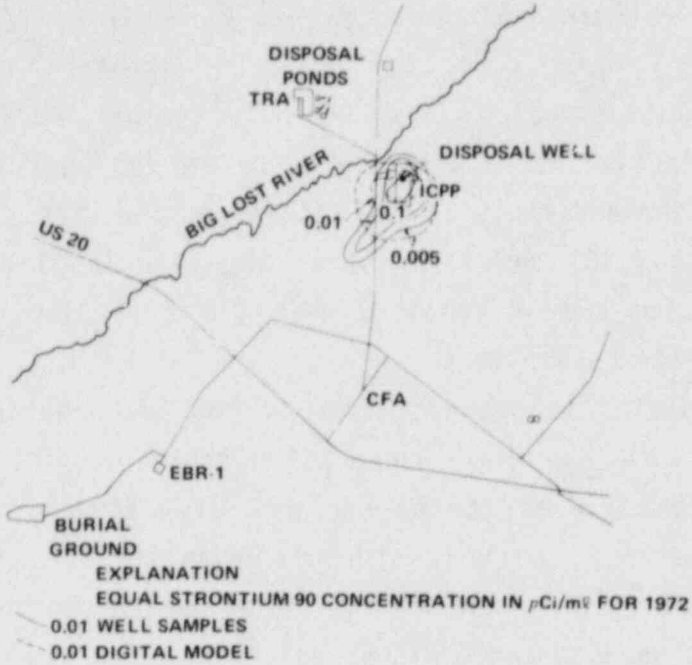


Figure II-27. Comparison of Computed and Measured ^{90}Sr Concentrations

<u>Source</u>: (Reference 669) J.B. Robertson, <u>Digital Modeling of Radioactive and Chemical Waste Transport in the Snake River Plateau Aquifer at the National Reactor Testing Station, Idaho</u>, IDO-22054, U.S. Geological Survey, Idaho Falls, ID, 1974

II.F.1.4.5 <u>Example 4: Waste-Package Performance Evaluation</u>

The fourth example shows how models are used in evaluating a particular component of a disposal system--the waste package.

An analysis of waste package performance is presently under way to determine the benefits of using long-lived waste packages that could release radionuclides only at low rates. The study predicts both the reduction in radiation doses delivered to people and the performance of packages of various designs (382, 656). Two possible but unlikely release scenarios are being considered for four host rocks (salt, basalt, granite, and shale). The first scenario involves ground water intrusion followed by radionuclide transport to a surface water body, and the second involves ground water intrusion followed by transport to a domestic well. Calculations of the performance of the entire disposal system are being made for combinations of disposal-system parameters that span the ranges present in real systems (water velocities, path lengths, dispersion coefficients, package container lifetimes, package release rates, and sorption coefficients). The dose-reduction calculations have been completed only for salt, but the package-performance calculations have been completed for all four media and for nine package concepts. The following conclusions can be made from the results of that study to date:

1. According to the far-field analyses of both scenarios at a salt repository holding spent fuel, waste-package components that either delay the initiation of release or control the release rate after initiation would be completely redundant in limited releases below those specified by Objective 2 in Section II.A.1.

2. Several package concepts are available that can delay the initiation of releases for hundreds of years.

3. The use of waste packages in the disposal system will provide an element of conservatism in disposal-system design.

II.F.1.5 Conclusions

The development and verification of models of single phenomena required for analyzing the long-term performance of mined geologic disposal systems are well advanced, and these models can now be routinely used. Even so, the development and particularly the verification are continuing. The development and verification of models that analyze several phenomena together are moderately advanced, and some of these models can now be routinely applied.

Laboratory, bench-scale and in situ tests are under way and/or planned (II.F.2) to assist in verifying modeling predictions.

The use of these continually improving models, along with the continually improving body of experimental data, will permit the performance assessments to be done more completely and with increased confidence. These assessments will be important in site selection, system design, and licensing.

The models have been used to assess the performance of disposal systems and have demonstrated that they can treat the important phenomena. Because the development and verification of models and the gathering of data describing sites are incomplete, these assessments have used conservative data derived from laboratory and field measurements. They have demonstrated that the models have been developed sufficiently for use with complex systems.

These studies have also predicted the consequences of releases of radionuclides from repositories in the far future. The vast majority of possible disposal-system conditions would not deliver any measurable doses to people. Some possible but unlikely phenomena, such as ground water flow directly through repositories, could deliver radiation doses that would be a fraction of the doses delivered by natural background radiation. The only studies that have predicted large doses to people have been based on what seem to be unrealistic assumptions, such as the occurrence of highly unlikely breaching phenomena or the omission of engineered barriers from the repository system.

The analyses performed to date give no indication that a mined geologic disposal system, designed and constructed according to the requirements described in this Statement, cannot isolate radioactive waste safely.

II.F.2 Experimental Basis for Model Development

Section II.F.1 discusses the models used to evaluate the performance of a mined geologic disposal system. Particular models discussed in that section are used to address specific components of the system--the waste package, the repository, the geosphere, and the biosphere. Integrated models will be used to analyze the entire system. An important part of the development and verification of each of the models is a methodical correlation with experimental evidence. The models for performance assessment in the NWTS Program are supported by a broad-based experimental program as described in the DOE/USGS Earth Science Technical Plan (203). These experimental programs are needed in all the steps of model development and application; however, they are emphasized in the following areas:

1. Model Development and Formulation. At the outset of model development for a particular system, some phenomena affecting the system may not be known. Scoping experiments performed to identify the important effects under the anticipated conditions should identify the potentially important phenomena. Those phenomena are subsequently incorporated into the model formulation.

2. Development of Data Base for Model Application. Application of models to a waste disposal system requires the incorporation of data describing the characteristics of either a generic or a site-specific system. These data are provided by experimental measurements on specific components of the system.

3. Model Verification. As models evolve, experiments are performed to evaluate the applicability of the models. Comparisons between model predictions and observed results are used to refine the models, if necessary, and to verify their predictive capabilities (see II.F.1.1.1).

In addition to supporting the development of models, experiments sometimes serve as demonstrations as well as scientific investigations. These demonstrations are culminations of specific steps in the program. They build confidence in the technical basis of the program and demonstrate initial steps in the conservative approach of the NWTS Program.

The experimental program of the NWTS Program refers to three general classes of experimentation: (i) laboratory tests, (ii) in situ tests, and (iii) observations of natural systems. Laboratory tests are sometimes further subdivided into laboratory-scale and bench-scale experiments, which typically focus on systems with characteristic dimensions of a few tens of centimeters and a few meters, respectively. In situ tests generally follow laboratory tests; and may involve expansions of time and spatial scales. In situ tests are performed under more realistic conditions than those possible in the laboratory, but control of contributory variables is more difficult. As a result, many phenomena are best investigated with laboratory tests, which provide more precision and control over experimental variables. Conversely, certain effects, such as those due to perturbations of the in situ stress state by excavation, are best studied by means of in situ tests. Several observations of natural systems are under way to obtain correlations which could not be obtained using the other investigative methods due to the long time frames involved.

This section summarizes the role and status of the Department's experimental program. Section II.F.2.1 discusses the laboratory testing program, relying heavily on references given earlier in the document. Section II.F.2.2 provides an overview of the in situ testing performed to date and briefly presents future plans for additional tests.

II.F.2.1 Laboratory Experiments

The models used to assess the performance of a mined geologic system rely on a wide range of laboratory tests to characterize the phenomena described by the models. Many of these tests were described earlier in this Statement; hence, this section outlines the principal activities from past and present programs within the NWTS Program and groups them according to the categories of models discussed in II.F.1.

II.F.2.1.1 Waste-Package Experiments

Widely varying experiments have been performed and are currently in progress to address interactions between the waste package and the host rock. These experiments principally address the quantity of fluids that could interact with the waste package, and the degradation of the waste package under interactions at repository temperatures. Studies of the effects of radiation on host rock and fluids have also been performed to investigate the impact on the waste package.

Results of laboratory experiments at both laboratory and bench scale for brine migration in salt were discussed in II.E.2. These results have enabled model developers to identify the phenomena that can influence the release and migration of fluids in salts. The information is now being used to formulate mathematical descriptions that correlate the temperature, temperature gradients, and stress state with fluid-migration rates (see II.E.2).

The behavior of fluids in basalt (631) and granite (715) is being characterized to describe the fluid chemistry and conditions that packages will encounter.

The testing of waste-package components in solutions representing the various host rocks is described in II.E.1. These studies provide information that will be included in the source term for the scenarios which are used to evaluate the consequence of radionuclide releases, and include the following activities:

1. Measurements of radionuclide release from spent fuel in various leachants at temperatures up to 300°C.

2. Corrosion tests of potential canister-sleeve and overpack materials.

3. Determination of radionuclide sorption characteristics and physical properties of candidate emplacement-hole backfill materials.

4. Corollary investigations of radionuclide speciation for various conditions induced by fluids in the host rock, radiation interactions, and the presence of other waste-package components.

As noted in II.E.1, sufficient laboratory experimentation and modeling have been performed on waste-package components that more complex experiments incorporating multiple components are now being designed. These experiments, involving both bench-scale and in situ tests, will investigate the effects of interactions (e.g., absorption of radiation and dissolution) among the waste form, the canister, and the backfill materials.

Data from the tests are incorporated into models which characterize interactions between package components and the host rock so that predictions of degradation for various fluid conditions can be made. The sensitivity studies discussed in II.F.1.1.4 are used to assess the dependence of the calculated consequences on various assumptions concerning waste-package integrity.

II.F.2.1.2 Experimental Studies to Support Repository Analyses and Design

Analyses of the pository performance during the operational and long-term phases include the use of models that describe the deformations, temperatures, and stresses in rocks, and the ability of rocks to transmit fluids. These models (described in II.F.1.3.2) use constitutive laws that describe the behavior of various rock materials. Constitutive laws are, in part, derived from observations on samples of rock taken from sites under investigation and then heated and deformed by laboratory testing. For example, creep testing (716, 717) is performed for the purpose of characterizing the primary, secondary, and tertiary creep behavior of rock salt. Similar experiments are being performed on basalt and granite to determine elastic properties, nonelastic stress-deformation behavior, and the influence of natural fractures.

In addition to the laboratory experiments using small samples, bench-scale experiments have been performed and are under way for both salt and granite (718, 719). Results from these experiments are compared with those results from laboratory-scale sample testing to determine the effects of scale on the material properties and phenomena of interest.

The relationship of these experiments to the characterization of a host rock is discussed in Chapter II.D. In general, they serve to build the data base for the modeling used to compare the predicted response of the repository configuration to the performance objectives discussed in II.E.2.

The evaluation and verification process for a specific set of these models, e.g., those for rock mechanics, is traced from laboratory testing, through bench-scale testing, and into field, or in situ, testing, in Figure II-28.

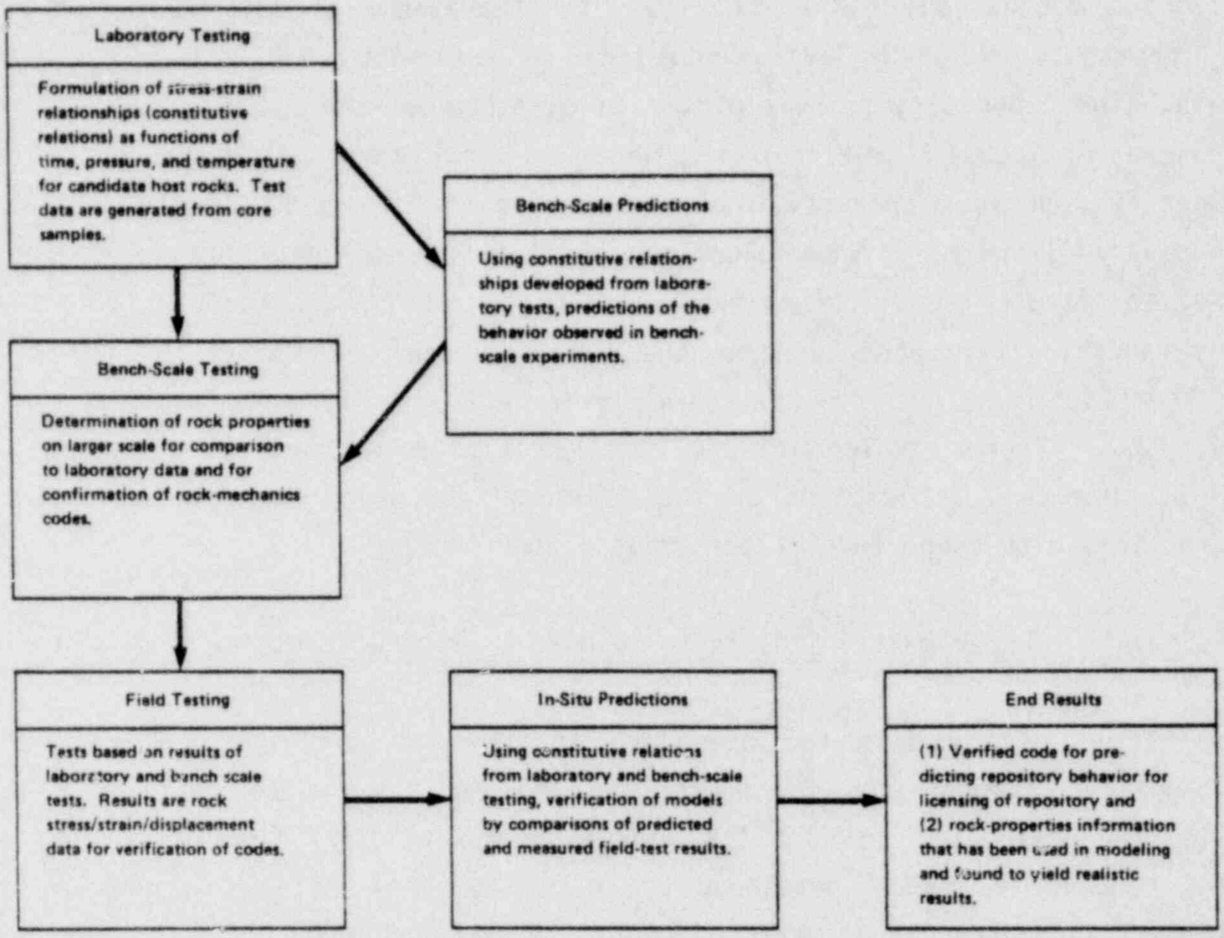


Figure II-28. Steps in Verification of Rock Mechanics Models

The assumptions about repository integrity that are made in the development of scenarios discussed in II.F.1.2 also are being evaluated by experiments addressed to repository sealing and backfilling. Examples of these activities are the studies of permeability, porosity, and geochemical stability of sealing materials, as discussed in II.E.2.

II.F.2.1.3 Experimental Studies to Support Geosphere Transport Models

The description of the geosphere transport models in II.F.1 includes a discussion of the repository configuration, as well as the physical, chemical, and hydrologic properties of the host rock and surrounding strata. The laboratory tests employed to characterize the natural systems are discussed in II.D.4. Numerous properties are evaluated in these tests; an especially important property used in modeling the transport of contaminants through the geosphere is radionuclide sorption. Experiments are being performed to discern the different geochemical interactions and their relation to the retardation parameters used in the models. Numerous studies of distribution coefficients, K_d, for various fluids and rock types have been made (715, 720). These studies include the sensitivity of sorption to changes, such as chemical complexation of the radionuclides which might occur due to interactions with components of the ground water (721).

II.F.2.1.4 Experimental Studies to Support Biosphere Transport Models

The models for biosphere transport predict the movement of radionuclides through the biosphere and the radiation doses delivered to people. As discussed in II.F.1.2, the modeling used in the NWTS Program relies heavily on models developed for other applications; consequently, no additional laboratory experiments have been implemented by the NWTS Program to refine or provide data for these models.

II.F.2.2 In Situ Testing

In situ testing supports the laboratory experiments that identify many fundamental physical and chemical phenomena and establish many mathematical relationships. In situ experiments are generally conducted for two reasons: (i) to determine whether any important phenomena are scale-dependent and thus have not been identified in laboratory experiments; and (ii) to verify that mathematical relationships for phenomena, based upon understanding gained from experiments on small samples studied in the laboratory, are applicable to large-scale assemblies that do not necessarily satisfy all the conditions controlled in the laboratory. In designing a waste repository, it will be important to understand the range of phenomena from the micromechanisms of chemical reactions to the macroreactions of rock responding to heat.

In situ testing is one phase of a sequential research and development program. The main value of such tests is that they are performed under conditions that provide for the natural variation of the geologic environment. Such experiments provide:

1. Development of data for use in models that are specific to a medium or a site.
2. Determination of the relative importance of various phenomena under actual repository conditions.
3. Evaluation of the capabilities of models for predicting effects.
4. Enhancement of conditions that can accelerate the interactions, making the measured effects represent those that may occur beyond the anticipated repository operating lifetime.

The models discussed in Section II.F.1 are divided into two types: near-field and far-field models. The spatial scale and time period for an in situ test are comparable to those used by a near-field model to predict temperature, stresses, and deformation of the rock mass around a disposal room, along with chemical interactions among the waste package, ground water, and host rock.

Measurements of rock response to excavation and to introduction of heat support both the development of models to predict effects during and after repository operations and the compilation of empirical data for future operations. These models can also be used to assess the long-term mechanical behavior of the rock in response to thermal load, its subsequent relief, and subsidence.

Experimental study of interactions between the waste package and the host rock under actual emplacement conditions and accelerated test conditions provides information on waste-package degradation and radionuclide movement in the very near field. In addition, many special questions are best resolved by underground experimentation. These include the development of special techniques for locating fluid pockets, voids, or fracture patterns in the host rock; collecting data on ventilation system performance; the measurement of background radiation levels; and other factors related to repository operation.

Long-term models cannot be directly evaluated at full scale under real conditions, since the scenarios assumed to generate serious consequences are usually beyond man's ability to create, and the time scale for the phenomena of interest is too long to accumulate data. However, components of these models, such as flow in aquifer systems, radionuclide sorption in rocks, source term conditions, and temperatures and structural deformation from heating, can be evaluated with in situ experiments of finite duration.

In situ tests also provide demonstrations that enhance the awareness, by both the public and scientific peers, of the development of waste management technology. Thus tests are important elements in the step-by-step process and illustrate the conservative approach endorsed by the Department. These full-scale demonstrations can also be used to gain experience with "live" operations that accumulate data on repository design and operational procedures.

Some in situ testing activities that address issues of spent fuel disposal have been completed, some are in progress, and more will be conducted as programs continue. Table II-15 is a compilation of these activities, which are discussed in the remainder of this section. In addition to the testing activities shown in Table II-15, a number of field tests are being carried out within the NWTS Program at various locations associated with potential sites. These field tests are presented in Chapter II.D and in Appendix B, and are not repeated here.

Table II-15. In Situ Testing Activities

Host Rock	Location
Salt	Lyons, Kansas (Project Salt Vault--1965-1967) Federal Republic of Germany (Asse II) Southeastern New Mexico (proposed) Avery Island, Louisiana
Granite	Sweden (Stripa Mine) Sweden (Studsvik) Nevada Test Site (Climax stock) Idaho Springs, Colorado Cornwall, England
Shale	Oak Ridge, Tennessee (Conasauga shale) Nevada Test Site (Eleana argillite)
Basalt	Hanford, Washington (Near Surface Test Facility)
Tuff	Nevada Test Site
Clay	Mol, Belgium

II.F.2.2.1 Lyons, Kansas (Project Salt Vault)

Project Salt Vault (116) was designed to establish the feasibility of and the techniques for the safe disposal of high-level waste in a bedded salt formation. The site was an abandoned salt mine near Lyons, Kansas, which had been operated by the Carey Salt Company. These tests, which were started in 1965 and ended in 1967, involved the emplacement of spent reactor fuel taken from the Engineering Test Reactor and electric heaters designed to simulate the thermal effects of the fuel. Seven canisters containing a total of 14 spent fuel assemblies were used in these experiments. The spent fuel was retrieved and returned to the Idaho Chemical Processing Plant (ICPP) at the conclusion of the tests. The objectives of Project Salt Vault were:

1. Demonstration of waste-handling techniques.

2. Determination of gross effects of radiation.

3. Determination of radiolytic production of chlorine.

4. Collection of data on creep of salt at elevated temperatures (100°C-200°C).

To achieve these objectives a series of well-instrumented experiments was carried out. The fuel canisters were emplaced in a hexagonal array--one canister in the center and the remainder at points on 1.5-m centers. These canisters were exchanged for fresh ones every 6 months over a period of 2 years. An identical array of electric heaters was installed as a control measure so that the contribution of radiation to the behavior of the salt could be determined. Also, one mine pillar was heated by a number of heaters placed in the floor around its base in order to provide additional information on the deformation of salt at elevated temperatures. The experimental measurements included salt temperatures, salt radiation doses, radiolytic gas production, thermal expansion of floor and pillars, and changes in stress and strain in pillars and other parts of the formation. Thermocouples were installed for the temperature measurements; surface and internal strain gauges were emplaced to measure thermal expansion and inelastic deformation; a portable extensometer gauge was used to measure roof-to-floor displacement; and leveling points were used to measure the thermal expansion of the floor. Gas was sampled from the annulus between the salt and an outer canister in the spent fuel array and in the heater array.

The experiments conducted under Project Salt Vault established empirically the magnitude and relative importance of the phenomena that would have to be considered in designing a repository. In addition, sufficient data were collected to allow for the development of mathematical models. The significant results from this project and supporting investigations were:

1. The observation that brine in the salt migrated to the heat source (116).

2. The determination of amounts and form of stored energy due to gamma radiation in salt (475, 722).

3. The development of a constitutive relationship for time-dependent deformation in salt (723, 724).

4. The refinement of a model to predict the distribution of temperature in the salt around the heat source (116).

5. The determination that chlorine gas generation from gamma radiation was insignificant (116).

II.F.2.2.2 Federal Republic of Germany (Asse II)

Asse II is an experimental radioactive-waste-disposal facility operated for the Federal Republic of Germany (FRG). It is located in an abandoned and redeveloped salt and potash mine in the small salt dome called Asse. This dome, one of the more than a hundred domes in the area, is near the frontier of the Democratic Republic of Germany. Asse II has operated as an experimental facility for the last 13 years (725).

Waste emplacement has involved almost all the low-activity wastes produced by hospitals, laboratories, and research facilities in the FRG. These wastes are packed in 200-liter drums (some with additional shielding), and the drums are subsequently piled in mined rooms. In addition, sample drums have been immersed in brine in remote parts of the mine in order to examine some effects of corrosion. Until recently, intermediate-activity wastes, which must be handled remotely because of the radiation output, have been accepted.* Emplacement is done in a large room dedicated to receipt of remotely handled intermediate-activity waste.

Hydrologic investigations around the salt dome and in the Asse hills have defined the general movement of fluids in the surrounding aquifers. Specific surface sources of brine have been investigated and the sources accounted for.

---

*Recently, receipt of both low- and intermediate-activity waste has halted, pending receipt of a license from the FRG Regulatory Office to cover this activity.

Asse II has been a functioning experimental repository with waste of intermediate and high activity. The repository has operated successfully since 1967, and so far no technical surprises have appeared. The local hydrologic system has been investigated, and the results of those studies will be used in long-term safety analyses (726).

The experiments in progress and being designed for the Asse Mine include (727):

1. Development and testing of disposal methods for solidified glass blocks of high-level waste, which includes closure measurements of a 0.3-m diameter borehole over a depth of 300 m, and the emplacement and later retrieval of the glass blocks from the storage boreholes.

2. Testing stability of underground openings at ambient temperature, including measurements of closure in both old and newly mined rooms and in an ellipsoidal cavity (10,000 m^3 volume) situated over a depth of 959 to 996 m, and of seismic activity within and outside the mine workings.

3. Emplacement of 2.75-kW electric heaters in a triangular array of drill holes in the floor for measurement of temperature perturbations in the salt, and of five 1.8-kW heaters in a single drill hole for measurements of drill hole closure and temperature perturbations and brine migration in the salt.

4. Development and testing of backfilling and sealing materials and techniques, using intervals of hot bitumen and a blended mixture of cement salt/gravel in a borehole, and alternating layers of fly ash and magnesium chloride-saturated brine in tunnels with small cross-sections.

The results that have come from in situ experiments conducted at Asse II have included the determination that paint on 200-liter steel drums immersed in brine pools has prevented the drums from corroding and agreement within 10% between a comparision of model calculations and in situ measurements of temperature for the heater tests.

The United States is currently developing a new Bilateral Agreement (728) with the the FRG to continue to exchange waste management information and initiate joint field testing projects in the Asse facility. Recommendations have been developed for areas of cooperation between the United States and FRG for the next 5 years (729). A joint brine migration test in Asse is planned, to provide verification of laboratory data and calculational models. Plans are also being exchanged on potential joint in situ tests in the area of waste/rock interactions.

II.F.2.2.3 Field Tests in Southeastern New Mexico

A repository, designated the Waste Isolation Pilot Plant (WIPP), was proposed for construction in bedded salt in southeastern New Mexico. It was designed for the disposal of transuranic waste from defense programs. In addition, it would have provided a location for in situ experiments with defense high-level waste. This program, designed to gain information about the impacts of heat and radiation in salt, would have been applicable to the disposal of spent fuel.

In his 12 February 1980 statement, the President concluded that the WIPP project, as currently authorized, should be canceled. The site near Carlsbad, New Mexico, will continue to be evaluated as a candidate site for a repository in bedded salt. If the Department elects to proceed with an early exploration shaft at the Los Medanos site, the experimental program outlined here could be implemented to support the commercial nuclear waste management program.

The research program at the Sandia National Laboratories, which supported the development of the WIPP, identified three phases for experiment activities:

1. Laboratory and bench-scale experiments that support model development and limited cooperative experiments conducted in existing mines.

2. In situ experiments associated with an early exploration shaft at the WIPP site and conducted without radioactive sources or waste forms.

3. In situ experiments with radioactive sources or actual waste forms.

Various laboratory and bench-scale experiments have been performed since 1976. In situ tests of borehole seals and evaporite gas permeability were performed in 1979 (730, 731). Tests of salt permeabilities at 2,100 feet below the surface were performed with a high-precision tandem packer system that measured gas permeabilities of approximately 1 microdarcy. The results of borehole-seal tests are reported in II.D.2.b. Cooperative experiments with local potash mines have been performed to determine the environmental conditions in the mines and the deformation of pillars during mining operations (732, 733). The pillar deformation measurements and associated instrumentation development (734) were performed during mining operations in which extraction ratios of 90% were used, resulting in complete room closure in a few weeks.

A set of in situ experiments was developed to be performed when site-exploration shafts were sunk to potential repository levels. The plan for these experiments (735, 736) included:

1. Repository design confirmation experiments that investigate the stability of various excavation configurations.

2. Heater tests to assess thermomechanical response of floors and pillars, investigate fluid migration, and determine performance of waste package components.

3. Experiments, related to operation and design, to investigate emplacement techniques, retrievability methods, mine-face scanning, and associated instrumentation development.

Plans for these experiments (735) include placement of both full-size waste canisters and unpackaged samples of various forms of bare waste directly into the salt.

II.F.2.2.4 Avery Island Field Experiments

Field experiments are currently being conducted in a test facility at the 168-m (550-ft) level of the Avery Island mine in domal salt. The mine is located near New Iberia, Louisiana (604).

At Avery Island, a series of experiments is in progress for examination of thermomechanical response and brine migration in domal salt, and for evaluation of experimental-equipment design and instrumentation system performance. In addition, experiments are planned for extended salt permeability measurements and for preliminary assessments of backfill consolidation, accelerated borehole closure, and fracture healing. These experiments will:

1. Evaluate the response of domal salt to a source of heat and compare this response to that of bedded salt as observed in Project Salt Vault (608).

2. Examine movement of synthetic brine in a temperature field and develop methods for future fluid-migration experiments.

3. Evaluate the performance of a new heater design.

4. Evaluate radar techniques for detecting structural discontinuities and other anomalies within intact salt and for locating waste canisters.

5. Determine the permeability of heated and unheated salt.

6. Evaluate borehole closure, backfill compaction, and fracture healing.

Three separate tests using electrical heaters in the floor of the mine have been in operation since June 1978. Experimental data are being gathered on salt temperatures and thermally induced displacements and stresses at different heater power levels, with and without a crushed-salt backfill between a protective sleeve surrounding the heater and the walls of the emplacement borehole. Measurements of mine environment conditions, borehole closure, and the generation of gas by corrosion are also being taken. Metal samples are emplaced to evaluate the corrosion characteristics of candidate canister and sleeve material (455).

Brine-migration experiments are being conducted at three positions. One position has no heater and assesses any natural movement. The other two include 1-kW heaters and involve monitoring of the migration of natural fluids and synthetic brine tagged with deuterium (737, 738).

A radiant-heater experiment is currently in progress. This experiment uses quartz lamps with a power capacity of 2 kW and the capability to raise the temperature of the salt to 200°C. A dry nitrogen and desiccant system is also incorporated to monitor fluid influx to the heater. A radar system under development at Sandia Laboratories has been used to scan the salt pillars to locate simulated waste packages (739, 740).

Current plans for additional experiments include permeability measurements in the natural salt and around heaters, measurements of the degree of consolidation of backfill at different temperatures and pressures, and an evalution of fracture healing in salt.

Results to date for the 1-kW heater tests do not show any major differences from those of Project Salt Vault. Measured values of thermal conductivity have been only about 10% higher than those measured in laboratory tests, probably due to the level of thermally induced stress in the salt.

Preliminary results from the radar experiments indicate that the interface between a simulated waste package and the salt could be identified with a spatial resolution on the order of 10 cm. Radar transmission through 180 feet of salt experienced little signal attenuation. Reflected radiation detected a 2.5 cm diameter steel pipe situated at a depth of 11 m into a salt pillar. The reflected signal was relatively strong, indicating that interfaces between salt and electrically conducting material can be detected in dome salt (741, 742).

II.F.2.2.5 Experiments at Stripa, Sweden

A cooperative United States-Sweden experimental program (743) to support radioactive waste management programs is being conducted at the Stripa Mine in Sweden. The experimental program involves a series of in situ tests conducted in a granite formation. The granite formation is accessed at a depth of 350 m through an iron ore mine that had recently completed its commercial operation. The Swedish-American Cooperative Program was initiated formally in July 1977. The principals are Sevensk Karbransleforsorjning (SKBF) and the Lawrence Berkeley Laboratory (LBL).

The experimental program is broadly based and addresses a wide range of questions. The tests of greatest interest to understanding the ability of hard crystalline rock to isolate the waste are:

1. Confirmation that existing computer codes are capable of accurately predicting the temperature profiles in a wet, jointed hard rock.

2. Measurement of the rock-mass permeability as a function of temperature and pressure.

3. Determination of the magnitudes and principal directions of in situ stress in the rock mass.

4. Determination of thermally induced stresses and deformations in the rock mass around electrical heater emplacements, and of any related phenomena due to heating of the rock.

5. Testing of a macroscopic method to define the combined bulk and fracture permeability of the rock.

The program conducted at Stripa is the first comprehensive set of in situ tests to evaluate hard crystalline rock as a medium for disposing of radioactive waste. The significant results to date of this set of experiments are given as follows:

1. Two different experiments using emplaced heaters have demonstrated that existing computer models can calculate accurately the temperature profiles in the rock. The experiments demonstrate that the predictions are accurate over a period of 20 years (this prediction results from the ability of one of the tests to compress 20 years of heat flow into 2) (744).

2. The measurements of the permeability of the rock mass indicated that the permeability decreased with increasing rock temperature. Other measurements showed that the permeability was independent of pressure (745).

3. The measurement of in situ stress in the hard rock showed that there was substantial scatter in the magnitude and direction of the stress (746).

4. The calculation of the deformation resulting from heating of the rock mass was greater by a factor of three than that measured in the experiments. Three potential causes for the discrepancies currently being investigated are (a) the validity of the input data; (b) the factors considered in the thermomechanical model; and (c) the adequacy of the instrumentation for the measurements (744).

5. The calculation of the change in stress resulting from heating of the rock mass agreed closely with the measurements (744).

6. The decrepitation on the wall of the hole in which the heater was emplaced increased with increasing temperature. The major spalling occurred when the wall temperature exceeded 300°C. Two separate phenomena are suspected of being responsible for the spalling, one that is time-dependent and a second that is stress-related. Agreement has not been reached on this issue (747).

The experiment to evaluate the fracture permeability is still under way. Conclusions regarding its capability of determining an integrated measure of bulk and fracture permeability will not be made for at least another year.

II.F.2.2.6 Experiments at Studsvik, Sweden

At Studsvik, about 90 km south of Stockholm, an experimental site in migmatitic-gneissic granite was chosen for in situ experiments on nuclide migration in fractured crystalline rocks (748). The fracture system appearing at the surface was mapped. Horizontal fractures were observed in core samples. Many fractures were filled with calcite and chlorite.

Boreholes were drilled in the test area, and different borehole investigations were carried out in order to determine the hydraulic conditions and to find flow points suitable for transport experiments. Three boreholes were selected for injection experiments.

Small amounts of Se, Tc, Sn, Cs, I, Nd, Sr, and Br were dissolved in ground water pumped from one borehole and injected at a specific depth in another. The nuclides were then forced into the fractures by continued pumping. Effective sorption coefficients were determined by these experiments and compared with those obtained from laboratory experiments that had been done for planning purposes. Overall, the in situ sorption coefficients showed good correlation with the laboratory values.

The earlier experiments were repeated after injection of bentonite slurry into the zone in which nuclides had previously been injected. The object of this series of tests was to determine how effectively the bentonite filler in the fractures would retard the release of radionuclides. Approximately 2 tons of bentonite were injected in this experiment. Radionuclides were injected into the bottom of the borehole. Two tracers, ^{82}Br and ^{85}Sr, were used to monitor the injection procedure. All the tracer activity remained in the borehole and could not be detected in the other boreholes after a period of approximately 1 year. The experimental results suggest that after bentonite injection the migration of Sr was less than 1/100 of the earlier migration.

II.F.2.2.7 Climax Spent-Fuel Test, Nevada

A test facility is being constructed at the Nevada Test Site (NTS), north of Las Vegas, Nevada, to investigate the suitability of granite as a host rock (749, 750). The two main objectives are:

1. To simulate the environment produced by a waste canister in a granite formation and to measure its effects.
2. To evaluate the differences between the effects of radioactive waste sources and electrical simulators on the granite.

Secondary technical objectives of the experiment are:

1. To compare the mechanical response of the rock formation to mining with its response to thermally induced loads.
2. To compare predicted and measured heat removed by mine ventilation.
3. To compare the thermomechanical response of fractured rock with that of relatively less fractured rock.

The test will be conducted in a representative granite that is composed of two main units (granodiorite and quartz monzonite) located on the northern end of the Nevada Test Site. The level at which the test facility is located is 420 m below the surface in a newly constructed tunnel adjacent to an existing set of tunnels. Eleven spent-fuel elements and six electrical heaters will be emplaced in a linear array on 3-m centers. Each fuel element will generate approximately 2 kW of thermal energy when it is emplaced, and this heat rate will decrease to 1 kW after 5 years. The thermal output of the heaters will be varied to match that of the spent-fuel elements.

The experiment was designed to simulate a repository with an initial loading of spent fuel at 44 W/m^2 (178 kW/acre) with drifts spaced on 15-m centers. Such a repository in granite would experience the maximum temperature in the rock approximately 50 years after the spent fuel is emplaced. In this experiment a similar maximum temperature in the rock adjacent to the emplacement hole will be attained within 2 years after emplacement. This temperature profile could be maintained in the rock surrounding the emplacement hole for more than 50 years. This test will demonstrate empirically the potential impacts of the disposal of spent fuel in granite in the near field on a scale comparable with a real repository.

Preliminary heater experiments were conducted in the Climax granite to determine the thermal conductivity of the granite rock mass. These measurements indicated that the thermal-conductivity values for the granite rock mass measured by in situ methods were approximately 10% to 20% greater than for intact samples measured in the laboratory. The field experiments also indicated that the measurements across major fractures were about 10% greater than the measurements parallel to the major fractures in the rock mass. Measurement of the permeability of the granite to gas (dry nitrogen) showed a decrease in permeability with increasing rock temperature. This is interpreted to mean that the pathways by which a gas could escape through the rock decrease in size as rock temperatures increase.

The effects of radiation on granite will be studied. Gamma radiation is not expected to have any significant impact on the mechanical or physical properties of granite (751); however, the chemical change in the minerals due to radiation is yet to be determined. The effects will be determined by temperature measurements, visual inspection of the boreholes, and microanalytical work on rock samples taken before and after the exposure. The differences between the effects of the electrical simulators and of the radioactive sources will be derived from this data.

Measurements of deformation were made in the rock above the tunnel opening during the mining operation. The deformation of the rock at the same positions will be measured during the course of experiments with the spent fuel. These measurements are being made because experts in the field of underground structures in hard rock have suggested that normal mining operations (creating the opening in solid rock) have greater impact on the surrounding rock than does heat from the waste. This experiment is significant because it will empirically measure the relative severity of the impacts of mining and of heat on the rock.

Experiments will test the validity of calculations of the quantity of heat that can be removed by ventilation of the mine. These calculations will be made by models that consider heat flow in fractured rock.

Spent fuel will be emplaced in the near future. It is anticipated that a period of 6 months to a year will be required to accumulate sufficient data to allow a meaningful analysis. The analysis of the data from the measurement of rock displacement during the construction of the tunnel is under way. The measurement will be compared with supporting calculations performed using appropriate mathematical models.

II.F.2.2.8 Colorado School of Mines Experimental Mine, Thermomechanical Test Facility

The Colorado School of Mines (CSM) has driven a tunnel in its Experimental Mine for use in the NWTS Program (752, 753). The CSM Experimental Mine is located near Idaho Springs, Colorado, and is situated in granite gneiss. The test facility room is situated under approximately 300 feet of overburden. The objectives of the tests are:

1. To assess the effects of blasting on the rock mass.

2. To determine constitutive relationships for crystalline rock masses.

3. To evaluate the heated flat-jack test as a method for obtaining the mechanical properties of jointed rock masses for input to thermomechanical models.

Extensometers and leveling pins were installed during construction to monitor the rock-mass behavior. Permeability measurements were made in boreholes parallel to the tunnel as the excavation proceeded. Additional work will include measurements of in situ stress and a statistical evaluation of fracture parameters and permeability measurements.

II.F.2.2.9 Eleana Argillite Test, Nevada

A series of tests has been conducted at the Nevada Test Site to investigate argillite as a host rock (754-756). The objective of this test was twofold:

1. To determine the phenomena that would occur if a thermal source were emplaced in a hole in argillite.

2. To confirm the application of thermal models to the prediction of effects in argillite.

The test was conducted in the Eleana argillite formation in the Syncline Ridge section of the NTS. It used an electrical heater similar in size to a canister of high-level waste, emplaced in the argillite approximately 10 m below the surface. The heater was operated at a power level of 3.5 kW, to simulate the power produced by a canister of 10-year-old high-level waste of borosilicate glass with 30% fission products. The test was conducted for slightly over 3 months, and temperature and pressure were the major parameters measured.

Temperature data taken during the test confirmed that thermal properties measured in the laboratory, used in conjunction with a standard model for predicting temperature, accurately described the temperature field surrounding the heater. Small deviations from the predicted behavior at the beginning and at the end of the test were observed. These deviations from predicted values suggested that unexpected phenomena were occurring in the test.

The primary information derived from this experiment is that phenomena different from those found in other geologic media were identified. This information is essential preliminary input to the design of any sophisticated test to evaluate the suitability of Eleana argillite as a medium suitable for a repository. The first phenomenon discovered was an upward movement of the water in the argillite when the heat was initially applied. The second was mineral dewatering, which led to material shrinkage and opening of joints.

Several conclusions concerning argillite have been drawn from this experiment. They are as follows:

1. At temperatures near the boiling point of water, the thermal and mechanical responses of argillite are dominated by the effects of clay contraction.

2. Contraction causes the opening of pre-existing joints that results in fracture-permeability transport of steam and water within the 100°C isotherm.

3. As dewatering occurs, the in situ thermal conductivity decreases below the value before heat was applied.

4. When these phenomena are included in the model that describes the temperature around the heater, accurate predictions of temperatures are obtained.

5. Although models adequately predict the thermal and mechanical response of argillite, the models could be improved if they were expanded to include the anisotropic properties and existing jointing systems in argillite formations.

The discovery of change in physical characteristics of the rock due to water movement and dewatering during heating indicates that careful attention must be given to these phenomena in the design of a repository in argillite. The modeling of these effects in argillite for a depth of 1,000 m indicates that they might not be as severe there because of the greater lithostatic pressure. Additional experiments at an appropriate depth would be required to confirm this prediction.

II.F.2.2.10 Conasauga Near-Surface Heater Experiment

An in situ experiment was conducted at a surface site near Oak Ridge, Tennessee, to evaluate the response of a shale formation known as the Conasauga shale to the thermal load that simulates the heat production from a canister of high-level waste (757). Two heaters 0.3 m in diameter and 3 m in length were emplaced in holes approximately 15 m below the surface. They were operated at a power level of 10 kW initially, and later at 6 kW.

The major objectives of the tests were to:

1. Measure the in situ thermal conductivity and determine if it would change due to heating.
2. Determine if the primary mechanism for heat transfer for a very wet formation involved both convection and conduction.
3. Determine if the mechanical response of the rock resulted in a significant change in the permeability.
4. Determine if the heat produced any mineralogic changes in the rock.
5. Measure any variation of ground water composition resulting from heating.

The test was conducted for approximately 8 months. The information gained from the experiment is given as follows:

1. The thermal conductivity of the shale changed from preheating value of 2.3 W/m ^{o}C to a value of 1.5 W/m ^{o}C at the completion of heating. The lower value is associated with shale that had been dehydrated.
2. Heat transfer in the formation is primarily by conduction, although perceptible displacement of the isotherms is caused by steam in the region immediately around the top of the heater. The isotherms for temperatures below the boiling point were accurately predicted by conventional models.

3. The overall behavior of the heated shale formation was a modest thermal expansion despite the contraction associated with the dehydrated material.

4. The presence of even moderate confining pressure has a greater influence on the properties of the shale than its loss of water.

5. The chemical environment created in the shale during the test was strongly oxidizing.

6. The permeability of the shale decreased with increasing temperature.

II.F.2.2.11 Near-Surface Test Facility

The Hanford Site in Washington has been identified as a location at which a repository in basalt might be placed (758, 759). The program includes a Near-Surface Test Facility (NSTF) for conducting in situ tests. The NSTF crosses several basalt flows, and the test area is located in the Pomona Basalt flow (II.D.5) with an overburden of 100 m. This flow is similar to the Untanum Basalt flow under the Hanford Reservation, which is considered a potential candidate for a repository site. The objectives of the test are to measure thermomechanical properties, establish limits for thermal loadings, compare modeling predictions with field data, and evaluate the impacts of radiation.

Several jointed-block tests, sometimes referred to as heated flat-jack tests and similar to the one at the CSM Experimental Mine, will be conducted. The purpose of the tests is to study the response of the jointed basalt rock mass to various loads as a function of temperature and structural orientation. The measured mechanical and physical properties of the rock mass at various temperatures will be used as input data to calculate the deformations that will be measured in the full-scale heater test described below. It has been proposed that the mechanical and physical properties measured by this technique are more representative of the rock mass than laboratory data and will improve the accuracy of models to predict stress and displacement perturbations in the rock due to heating.

The first full-scale heater test involves the emplacement of an electric heater the size of a canister of high-level waste in a hole equal in size to that proposed in repository designs. The hole will be surrounded by peripheral heaters to control the temperature field and gradients. Instruments to measure the temperature of the rock, as well as displacement due to thermal expansion, will be employed. Full-scale and peripheral heaters will be operated at power levels that approximate the conditions in a conservatively designed repository. The main objective of this experiment is to test the predictive calculations for the displacement of hard rock under thermal load, and to evaluate a new method for obtaining mechanical and physical properties of the rock mass.

The second full-scale heater test has the same configuration as the first, but without peripheral heaters and instrumentation to measure rock displacement. This test will establish empirically the maximum power load that can be sustained in an emplacement hole in a basalt formation. This test will determine the maximum temperature that the inside of a waste emplacement hole can tolerate without compromising the retrievability of the waste.

A test using spent fuel will be undertaken to determine the combined effects of heat and radiation on the basalt and potential canister materials. In addition to spent fuel, a canister of high-level vitrified waste will be emplaced in the basalt. While basic knowledge of the impact of gamma radiation on crystalline solids indicates that there should be no significant impact on the mechanical or physical properties of the basalt, the impact on chemical stability remains to be established.

The first jointed-block test and the first heater test will begin in 1980. By mid-1981, it will be possible to evaluate whether thermomechanical models, using properties derived from the jointed-block test, will improve the accuracy of predictions of thermally, induced stress and deformation in hard rock. The second heater test will begin in 1981. It is anticipated that the determination of maximum temperatures and power levels for an emplacement hole for basalt will be determined by 1982. The emplacement of spent fuel and high-level vitrified waste will take place in 1982.

II.F.2.3 Observations of Natural Systems

The models discussed in II.F.1 simulate complete physical phenomena over time scales which, in many cases, are far too long to duplicate in laboratory or in situ experiments. A few cases in nature provide analogs of situations of interest to waste management; consequently, investigations have been initiated which are intended to extract corroborating data for the models. Some examples of current investigation and their role in supporting model development are given in the following sections.

II.F.2.3.1 Natural Reactor Fission Program

Field Studies are in progress at a naturally occurring deposit of radioactive wastes near a uranium mine (Oklo) in Gabon, West Africa (760). These wastes were created 2 billion years ago when sufficient uranium ore accumulated to sustain a fission chain reaction for several hundred thousand years.

These studies are expected to yield information on the release rate of waste products. Examples of the relevance of data collected to date on the release rates of radionuclides from spent fuel appear in II.E.1.5. Correlations of observed release rates in the field and commensurate laboratory experiments can be used to support specification of the source terms used in long-term performance assessment calculations.

In addition, these studies will be used for comparison with models for radionuclide transport. Current investigation includes the use of elements and isotopic analyses to locate transport paths for specific radionuclides. The quantification of distances traveled by these radionuclides from the natural reactor site will be performed and comparison will be made with transport calculations.

II.F.2.3.2 Study of the Morro do Ferro Thorium Deposit

An evaluation of radionuclide transport models is also being performed with measurements of the transport and dispersion of radionuclides from the Morro do Ferro thorium deposit in Minas Gerais, Brazil (761). Measurements of initial dissolution and subsequent movement of thorium, uranium, and other rare earths away from a well-defined ore body in water-saturated media will be used to calibrate models for plutonium transport.

II.F.2.3.3 Correlation with Uranium Leaching Experiments

A third example in which radionuclide transport models are being evaluated within the NWTS Program is the cooperative study by the USGS and U.S. Bureau of Mines of a uranium-leaching experiment near Casper, Wyoming (762). Data on uranium leaching and transport will be correlated with the models to assess the validity of source term assumptions and radionuclide retardation in ground water.

II.F.2.4 Experiment Summary

The development and verification of models to assess the performance of a mined geologic disposal system has incorporated a wide variety of experimental programs. This experimentation has been performed at various scales over the last 15 years and has resulted in the development of numerous models to predict the performance of the waste package, the repository, and the transport of radionuclides through the geosphere. These models have been supported by the adaptation of existing models from various technical fields such as hydrology, rock mechanics, and biosphere radionuclide transport, which have an extensive experimental basis. While modeling of individual phenomena is well developed, there is current emphasis within the NWTS Program on verifying models and providing site-specific data to evaluate designs at potential sites. The methodical application of experiments which successively incorporate the effects of time and spatial scale and the impact of synergistic effects of various phenomena will be combined with a priori predictions, scientific overviews, and quality assurance programs to further strengthen the technical basis for the models. This process is expected to provide models sufficiently refined to support the successful licensing of a mined geologic disposal system.

II.F.3 Operational Phase Safety Assessment

As indicated in Objective 3 in II.A.1, the operations at a repository should be conducted in such a manner that any risks are comparable to or less than those posed by existing nuclear fuel-cycle operations. Similarities between the designs, storage requirements, and handling requirements at repository sites and those at other fuel-cycle operations suggest that existing regulatory guidance from the Nuclear Regulatory Commission may be applicable to the NWTS rogram (763-771). Even some aspects of those activities associated with shaft and disposal room construction, waste emplacement, and also subsequent backfilling which have not been previously addressed in Commission licensing may be covered by this guidance. This Section reviews the methods available for assessing the radiological safety of the operational phase activities and considers potential issues which may result from such activities.

The discussion presents an overview of the operations expected at a repository, followed by a discussion of normal operating considerations. Safety analyses of abnormal or accident situations, including the consideration of emergency response planning, waste retrieval, decommissioning, and post-closure monitoring are then addressed.

II.F.3.1 Operational Phase Activities

The operational phase of a repository is defined as the time from receipt of the first nuclear waste on site through repository closure and sealing. During this phase certain activities, such as the continued excavation of disposal rooms, waste emplacement, and the backfilling of disposal rooms after waste emplacement, may occur concurrently. The options related to such activities are discussed briefly in II.F.3.2.

The major safety-related steps during the operational phase are:

1. Receipt and Storage--The on-site transportation, receipt and storage of spent fuel will be conducted in a manner similar to that currently practiced at existing nuclear installations.

2. Packaging--Spent fuel assemblies will be removed from the storage basin, taken to a hot cell, and placed in canisters and sealed. Stabilizers (see II.E.1) may be added to the canisters, if appropriate, and special packaging of damaged fuel assemblies may be provided if necessary.

3. Transfer--Sealed canisters will be taken from the hot cell and transferred to the underground via the waste shaft hoist system. During such operations appropriate radiation shielding will be provided.

4. Underground Transport and Emplacement--Canisters will be transported in special shielded vehicles to the ultimate disposal location for emplacement. Emplacement may include placing sleeves and special backfill materials around canisters.

5. Closure--Upon completion of all waste emplacement activities, all disposal rooms and abandoned access and ventilation corridors will be backfilled. (Some backfilling of individual rooms and drifts may have occurred during the emplacement phase if approved by the Commission for specific facilities.) This phase includes sealing of the shafts and decommissioning.

The operational phase has been divided into these five steps to highlight the safety considerations. More detailed descriptions of the operating phase for a spent-fuel repository in bedded salt and a solidified high-level-waste repository in dome salt are contained in NWTS Program documents (116, 772-775).

II.F.3.2 Normal Operations

As briefly described above, the surface portion of the repository facility will be comparable to existing fuel-cycle facilities. Except for lower extraction ratios and differences due to usage, the below-grade repository structure will be similar to commercial mines that use room and pillar construction methods for the extraction of minerals. Thus, there will be a series of rooms separated by pillars that are large enough to provide the required stability (an extraction ratio of less than 20% is likely; see II.E.2). The following discussion briefly examines operational phase activities, methods of analysis, and requirements relating to the safety of normal operations.

It is instructive to note some of the differences in operational phase safety concerns between repositories and power reactors. In a power reactor, fuel assemblies are subjected to high operating pressures and temperatures and the fission product inventory within an assembly is at its maximum. The dynamic nature of reactor systems and large temperature differences necessitate numerous active protective and control systems. Conversely, spent fuel assemblies at a repository are at much lower temperatures and will not be subjected to the environmental conditions which are encountered in reactor operation. Consequently, repository operations will not need to contend with large internal driving forces that could lead to waste release, and they do not require the level of sophisticated active control systems associated with certain other fuel-cycle facilities.

On site the spent fuel will be transported, handled and stored in a manner similar to that described in the discussion of spent fuel handling in IV.D. Disassembly of fuel assemblies may occur for certain package configurations (II.E.1); for example, a process may be necessary for removing and separately packaging damaged fuel pins. Consequently, the potential for routine releases of radioactive materials is limited to those which are associated with possible surface contamination or gaseous leakage from some spent fuel assemblies.

Removal from storage and packaging will occur under controlled conditions. In addition to the incorporation of radiation shielding, systems for controlling potentially contaminated air and water will be provided. Water purification and air filtration will be employed to control such releases in accordance with applicable regulations. For normal operations after the spent fuel is placed in canisters, the potential for release of radioactive material will be small. Possible release points during normal operations can be identified by analyzing the operations taking place and estimates of the expected radiological source term can be made. The effectiveness of liquid and airborne treatment systems can be evaluated in the same manner as is the effectiveness of existing reactor and fuel-cycle facilities (764). The effectiveness of treatment systems will be judged by assessing performance based on either existing or forthcoming standards. Specifically, the Environmental

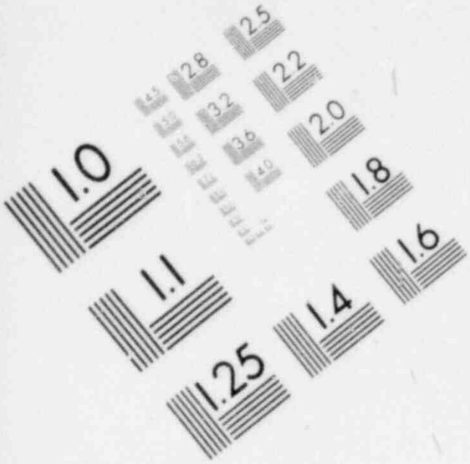
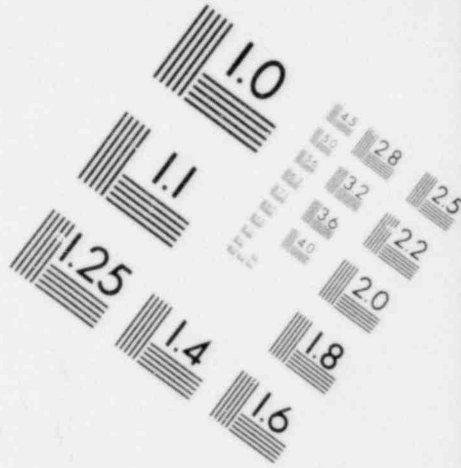
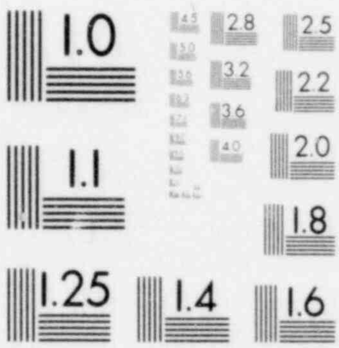
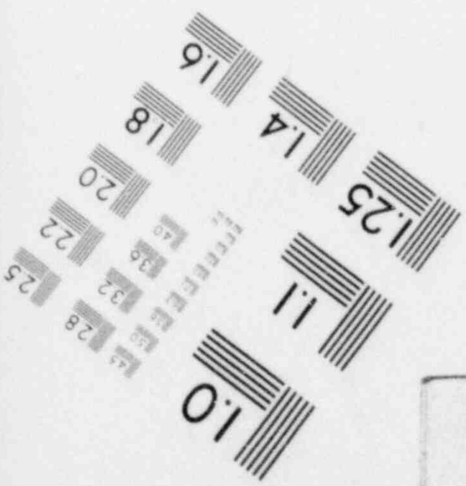
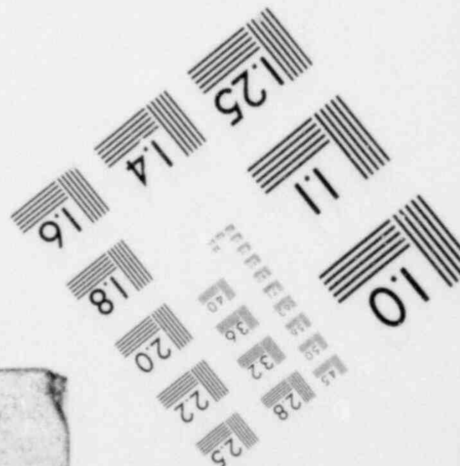



IMAGE EVALUATION
TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART

Protection Agency has issued 40 CFR Part 190 as a standard for routine effluents from fuel-cycle facilities. A similar numerical standard for waste management support facilities has been planned by the EPA but has not yet been published. In the interim, the Department will focus on 40 CFR 190, as the forthcoming waste management standard is expected to be similar in nature for the repository operations phase.

Formal license conditions and/or technical specifications will be used to prepare operating procedures and limits. Operation of the supporting facilities and equipment in compliance with such conditions/specifications will help ensure that routine releases will be acceptably low. Examples of the levels of human exposure from these releases are presented in II.F.4.3.

II.F.3.2.1 Occupational Radiational Exposures

Preliminary calculations for estimating occupational radiation exposures have been performed, and desirable and necessary attributes of a health physics program have been identified (772). Occupational radiation exposures associated with the transfer and emplacement of waste below ground could be due to:

1. Direct radiation from the transfer cask during loading, transfer and emplacement.
2. Direct radiation from the emplaced canister through the shielding plug and cover.
3. Radiation transmitted through the host media.
4. Naturally occurring radiation (e.g., radon).
5. Radiation carried through the ventilation system.

The two classic controls on occupational exposures are: (i) limiting the length of time that workers remain in radiation fields or contaminated areas, and (ii) limiting the intensity of the radiation field and the amount of contamination, if any. It is common practice to segregate parts of facilities into radiation zones within which access is controlled (763, 764). Radiation shields and geometry considerations will be included in repository design to reduce radiation levels, and equipment designs and layouts will be planned to minimize potential exposure times.

For example, remote handling methods, if necessary, will be used for maintenance operations involving extended periods of work with contaminated equipment or equipment in areas of high radiation. The maintenance and repair of cranes and manipulators in hot cells, for example, should minimize the need for worker ingress.

The Department is committed to maintaining occupational exposures as low as reasonably achievable in its facilities (776). NRC regulations (10 CFR 20) and guides (770, 771) are available to provide guidance in this area.

During normal operations, excavation of additional disposal rooms, emplacement of the canisters in disposal rooms, and either backfilling or sealing of filled rooms will occur. To accommodate such activities without disrupting normal operations, the underground structure will be developed in such a manner that the waste canister transfer and emplacement operations will be physically separated from the other activities. In order to minimize the likelihood of accidents involving the canisters and to reduce occupational exposures, it is possible to provide several transfer corridors and to adopt a canister emplacement sequence involving use of the extremities of disposal rooms first.

II.F.3.3 Safety Analysis

Many operations in the surface facilities will be similar to operations currently ongoing in licensed facilities such as the General Electric-Morris, Illinois, installation (777). Many surface facility operations, such as dry handling of fuel or packaging, will employ technology similar to that previously reviewed by the Commission for operations at reprocessing facilities (778, 779) and laboratory hot cells. Operational phase risks can be partially assessed in terms of those Commission requirements which appear relevant. Issues such as emergency planning and accidents can be assessed in light of existing and proposed criteria and practice (e.g., 10 CFR 50 Appendix E, proposed 10 CFR Part 72, 10 CFR, Part 71, and proposed Appendix P to 10 CFR

50). Such requirements and proposals are directly applicable to receipt, storage, and packaging.

The requirements for activities conducted below grade in the repository structure, such as underground transport and waste emplacement, need to be evaluated as to their potential effects on long-term repository behavior. Maintenance of long-term repository integrity as related to excavation is addressed in II.E.2. During the operational-phase safety assessment, the design and procedures will be evaluated to ensure that long-term behavior of the repository will not be adversely affected by the operational-phase activities. Evaluations conducted to date have discovered no potentially disruptive events which could not be avoided or compensated for by design.

II.F.3.3.1 Methods of Safety Analysis

The methods to be used to identify and assess operational safety impacts will include the following:

1. Identification of potential accident initiators, determination of accident scenarios, and evaluation of proposed operating procedures for possible human error.
2. Accident analyses, including evaluation of potential impacts on repository effectiveness.
3. Comparison of calculated releases with applicable dose guidelines.
4. Based on the results of 2 and 3, modification of design and procedures to ensure compliance with applicable requirements.

To assist in this approach, fault and event trees may be applied to identify the various outcomes of a given initiating event. Subsequent scenario analysis will be performed by conventional techniques (i.e., heat transfer, stress analysis, structural analysis, etc.). Consequence analysis will be done similarly, using available state-of-the-art computer models, such as INREM-II, which incorporate the ICRP task group lung model (780, 781).

Events will be evaluated according to their impact on either (i) operations aboveground or belowground, or (ii) long-term performance. Based on their calculated consequences, the most limiting will be identified as design-basis events. Such events will be used to evaluate the adequacy of the design, operation, and specific procedures. Any potential operational-phase events shown to result in significant release or to have significant impacts on long-term performance will be prevented or mitigated by design, operating procedures, or technical specifications. The repository will be designed to ensure that the event is not credible or that the event results in acceptable consequences. Examples of such events and possible mitigating measures include:

1. Inadvertent Impact of Canister During Waste-Handling Process. Potential short-term releases of radiation due to possible puncture of a canister would be controlled by ventilation system isolation and/or by filtration. Increased local stress in the canister, which might increase stress corrosion rates or accelerate the degradation of the canister, would be mitigated by canister overpacking so as not to affect long-term isolation. Similarly, the canister design will include provision for impacts that may occur during handling (see II.E.1), or it could be returned to the surface facility for repackaging.

2. Unacceptable Waste Characteristics. Accepting waste with characteristics significantly different from those for which the repository was designed could adversely affect the handling operations and waste package performance. A quality assurance and surveillance program will be used to assure that spent fuel accepted for disposal will be in conformity with predetermined waste acceptance criteria.

3. Cask Drop. The failure of a shaft hoist or elevator could cause a cask to drop the length of the shaft. Unmitigated, such a drop might breach the cask and disrupt normal operations. In addition to a highly reliable hoist system, mitigation measures such as velocity-limiting devices in the shaft and/or impact absorbers in the cask or base of the shaft are being considered. Operating and maintenance procedures will further increase hoist reliability.

4. Floods. Surface water inflow through shaft openings, due to phenomena such as rain, flooding, dam failure, etc., is not expected to cause repository flooding. Siting criteria (e.g., avoidance of flood plains) and design criteria (e.g., protection against floods) will be employed. During repository construction and operation, protection against ground water inflow will be provided. During shaft construction, water inflow will be controlled by grouting or other means. Dewatering systems will be designed to handle and measure possible water ingress during operations.

II.F.3.3.2 Example Analysis

The following example highlights a specific hypothetical problem and notes mitigating measures that could be considered.

The reliability of shaft hoist systems is affected by the design and operational procedures. For example, the hoist system can incorporate redundant cables, each capable of performing the required functions, or include provisions such as discussed in item 3 above. While design and operating provisions should make waste hoist failure unlikely and impact absorbers can reduce disruption of a canister, hoist failure resulting in the drop of a spent-fuel canister under conditions similar to those envisioned for a commercial repository has been analyzed as a potentially limiting event (782). The ventilation system in the hoist shaft could be designed in such a way that, in the event of an accident, air flow could be stopped and/or treated by high efficiency particulate air (HEPA) filters. Such a design would mitigate the release of particulate matter. In order to consider the worst consequences, however, an analysis was performed that assumed that ventilation air flow would continue after the drop and be exhausted directly into the environment without filtration. (Radioactive particulate matter would not be readily dispersed to the site environs unless both the ventilation system were allowed to continue running and the filters failed to perform.) Such an analysis is useful in placing an upper bound on the relative off-site impacts of gaseous and particulate releases.

Without filtration, the controlling dose was calculated to be about 70 mrem to the bone of a receptor at a position 3-1/2 miles from the release point. If the ventilation air were treated by exhaust filters, significantly lower calculated doses would result and the critical organs affected would be different. For example, the controlling doses for the case with filtration would be about 0.01 mrem to the lung and 0.009 mrem to the bone. These dose commitment figures represent integrated effects 50 years after exposure and are small fractions of the doses received from natural background radiation.

The results available to date demonstrate that, in the case of a canister hoist drop, the consequences are comparable to those associated with postulated accidents at existing fuel-cycle facilities. Furthermore, the results are very minor compared to many reference exposure criteria. For example, dose guidelines used in NRC siting evaluations of nuclear facilities are specified in 10 CFR, Part 100, as 25 rem whole body and 300 rem thyroid. Dose guidelines suggested in the proposed 10 CFR, Part 72, are 5 rem whole body. Lung and bone dose guidelines are not specifically addressed in those guidelines, but equivalent organ doses reflecting a whole-body dose of 25 rem have been used in licensing reviews by the Commission staff. These equivalencies are based on dose limits suggested by the International Commission on Radiological Protection for specific organs (783). The resultant reference exposure criteria are: whole body, 25 rem; bone, 150 rem; and all other organs, 75 rem. These values are not necessarily intended to be used as acceptable dose limits for severe accidents at a repository, but they provide a benchmark by which margins of safety have been assessed in the past and establish an exposure ratio for organs of interest.

Systems will be designed such that, in the event of accidents, involuntary exposure of both workers and the general public will be minimized. A value of 5 rem whole body is in interim use in the NWTS Program. This corresponds to an upper limit protective action guide value for the general public issued by the Environmental Protection Agency (784).

II.F.3.4 Emergency Response Planning

Emergency response plans will be developed in conjunction with proper siting and security arrangements to mitigate the impacts of unforeseen disruptive events on the public health and safety. The Department is following the development of requirements for radiological emergency plans for reactors which are currently being revised by the Nuclear Regulatory Commission (785). Although evaluations of on-site activities conducted to date have not identified events which require off-site emergency reponse plans, the Department recognizes that incorporation of additional precautions for protecting public health and safety is a prudent step. Consequently, emergency response plans will be developed in conjunction with State and local governments before authorization of construction so that there will be predetermined actions that can be taken in the event of unforeseen industrial or radiological accidents associated with repository operations. These activities will be coordinated with the Federal Emergency Management Agency (FEMA) and the Nuclear Regulatory Commission.

II.F.3.5 Waste Emplacement and Retrieval Considerations

Confidence in the suitability of the repository will be high at the time waste emplacement operations commence because of extensive review within the NWTS Program, peer reviews, and Commission licensing reviews. The verification of the suitability of the repository system will continue, however, throughout the waste emplacement period. Waste emplacement operation will allow for testing and monitoring during the first few years of repository availability.

Retrieval refers to the removal of the canisters from the repository if the long-term performance of the repository system were found to be unacceptable at some point prior to sealing. The initial waste emplacement will be conducted in a deliberate manner to allow for testing and monitoring during the first few years of repository availability. Instrumentation will be installed with the initial canisters. The details of this monitoring program will be developed in conjunction with the Commission licensing review.

Such a program will allow for the collection of additional data that can aid in minimizing uncertainty and in further assessing the response of the geologic system to waste emplacement. The results of those evaluations will be compared to results from in situ testing during the subsurface construction period and throughout the operational phase of the repository to ensure site and design adequacy. From these types of activities, further data will continue to become available for use in assessing and verifying computational models that form the basis for long-term projections (II.F.1).

Design features will be provided to allow for the retrieval of emplaced canisters throughout the operating phase. Waste package design (II.E.1) and repository design (II.E.2) have been considered elsewhere in this statement. This section describes the present concept for retrieving wastes from the repository during the operational phase.

Conditions that could lead to a decision to retrieve emplaced waste have been considered. Three examples of these conditions can be outlined as follows:

1. The design of the repository will be based on data obtained and upon accepted thermal, mechanical, and hydrological models (II.F.1). As mentioned, conservative allowances will be made to accommodate design bases (II.A.2). Nevertheless, retrieval of the waste and abandonment of the repository could conceivably be required if tests and acquired data show that a sufficient degree of confidence could not be provided.

2. The testing program might show that certain waste packages have defects that could be corrected by retrieval, overpacking, and re-emplacement.

3. A portion of the repository might be found to be unsuitable, and the wastes could then be removed and re-emplaced in a different section of the repository.

The most difficult (and probably least likely) case would be imposition of a Type 1 condition (total waste retrieval) near the end of the operational phase. The repository would have been fully operational and, if permitted by Commission licensing requirements, certain waste disposal rooms would have been backfilled when filled. Some rooms could be only partially filled with waste and therefore not backfilled, some rooms could be filled but not yet backfilled, and some rooms could be filled of waste and backfilled. Toward the end of the operating period an increased number of rooms could be in the latter condition.

Retrieval probably will not be based on an immediate threat to the repository but rather on loss of confidence in long-term containment. If it were decided to completely remove all waste from the repository because of lack of confidence in the natural system, there would be no further need to preserve the long-term integrity of the host rock. In such a situation, additional shafts could, if necessary, be constructed to facilitate retrieval. However, if only a partial removal were decided upon, precautions would still be taken to preserve the integrity of the remainder of the repository.

Sufficient testing would have been conducted to identify the preferred method of handling retrieved wastes. The waste package design should be compatible with the retrieval concept, and prototype retrieval equipment and procedures should be field tested and proven. (Some experience has already been gained in spent-fuel retrieval. In the Lyons, Kansas experiment, (116), 14 assemblies in stainless steel canisters, filled with helium and hermetically sealed, were retrieved and removed off site after about 6 months underground.) Retrieval operations would continue over a period of years. The retrieval process would probably involve two major phases: removal of the waste from the repository, and disposition of the waste packages.

Retrieval of the waste from the repository would involve the following steps:

1. Re-excavation of the backfill from the disposal rooms, if required.
2. Removal of the packages from the disposal holes.
3. Transfer of the waste to the surface.
4. Repackaging of damaged canisters, as necessary.
5. Placement into an alternate storage location or disposal site.

Engineering difficulties posed by emplacement and retrieval of canisters has received consideration, and alternative retrieval methods have been studied (786, 787). Mishaps owing to operator error (e.g., damage of waste containers through contact with heavy equipment) will be considered as part of the retrieval planning effort. For efficiency, the procedure could begin by removing the waste from open rooms and simultaneously removing backfill from any closed rooms. A number of canisters could become externally contaminated or damaged by equipment. Such canisters would be transferred to the surface and processed through the packaging facility for repackaging or overpacking.

During retrieval operations, waste removed from the repository could be transferred to temporary surface storage rather than to another repository. This approach significantly reduces problems associated with waste transport. Ultimately the waste will either be transported and disposed of in another repository, or it will be repackaged for re-emplacement.

Using presently known storage techniques (see Part IV), construction would be rapid and uncomplicated. The need for additional storage space would be determined when the decision for retrieval is made.

In summary, the following points are pertinent to the consideration of retrieval:

1. Both limited and total retrieval are unlikely events, the latter being least likely.

2. Mechanisms and procedures for retrieval are being developed.

3. Design criteria will incorporate retrieval considerations.

4. Retrieval operations would not require immediate action. They could be accomplished in an orderly and planned manner with adequate time for construction of alternate storage facilities.

5. Waste retrieval would be primarily an operational health physics concern (operator protection). Risks to the public health and safety due to retrieval would be very low due to repository and waste package design.

II.F.3.6 Closure

After the design capacity of the repository is achieved, routine operation will be terminated. Although surveillance of the repository may continue for several years, the eventual disposition of the repository will need to satisfy the following overall objectives:

1. The above-ground portion of the site will be restored such that radiation or contamination levels are sufficiently low to permit release of the site for unrestricted use, from a radiological standpoint.*

2. The underground facilities will be sealed.

A decontamination and decommissioning plan for the surface facilities will be developed and submitted to the Commission when appropriate. Although regulations specifically applicable to a repository are not currently in existence, there is sufficient licensing experience with nuclear power plants and with fuel-cycle facilities (e.g., uranium mills and fuel fabrication plants) for interim guidance. Guidance on acceptable levels of surface contamination attributable to unrestricted release at a site also has been provided (788).

Although minimal radioactive contamination is expected, appropriate radiological surveys will be conducted. Based on early consideration of decontamination and decommissioning objectives during the design stage, most structures are expected to have little, if any, contamination. If necessary, chemical and mechanical clean-up will be conducted in order that conventional dismantlement techniques can be used. The resulting rubble and wastes will be either shipped off site or placed in the repository for disposal. Two examples of nuclear facilities which have been decommissioned and dismantled are the Elk River boiling water reactor at Elk River, Minnesota, and the Plutonium Fuel Facility at Pawling, New York.

---

*Consideration must be given for permanent markers (II.E.3) and for access control during post-closure R&D activities.

During the course of normal operations, some of the storage rooms may be backfilled. Decommissioning of the belowground facilities will result in the backfilling of all remaining storage rooms and corridors. In addition, the shafts will be sealed (sealing techniques are discussed in II.E.2). The specific sequence of activities to be undertaken will be developed as part of the decommissioning plan, which will require Commission approval.

Inasmuch as deep geologic repositories will be first-of-a-kind facilities, extensive analyses supported by testing and monitoring programs will be conducted. addition, continued monitoring of the site environment after closure is being contemplated as a prudent measure. Changes in the immediate repository environment (such as the temperature of the host rock) will have been be monitored as part of an R&D program. Projections and monitoring of temperture effects for longer time will be possible at the time of closure. Although changes in the general environment are not expected, increased scientific data would provide a better baseline for future generations. In conformity with Objective 6 (II.A.1), the performance of the repository system will not require active maintenance or surveillance for unreasonable periods in the future. It should be possible, however, for future generations to conduct monitoring programs at their discretion. Consequently, the Department is planning a series of surface measurement systems for possible application. Such systems will measure the behavior of the ground surface, ground motion, or local hydrology.

The desirability and need for such monitoring will be determined in consultation with local, State, and other Federal agencies.

II.F.3.7 Summary and Results

Based on the evaluations available to date, operational-phase activities do not appear to be a limiting factor for an acceptable repository. Methods required for safety analyses are currently available. Satisfactory design, construction, and operation can be achieved with no significant effect on long-term repository performance, and the risk of the repository to public health and safety can be shown to be comparable to that of existing licensed nuclear fuel cycle facilities. Emergency planning will be coordinated with State and local governments and other Federal agencies.

Many similarities exist between the operation of repositories and that of other facilities presently licensed by the Nuclear Regulatory Commission. Based on experience with those facilities, it is likely that repository systems can be designed and operated to meet Objectives 3, 4, 5, and 6 of Section II.A.1 during the operational phase. Existing analytical methods are, for the most part, sufficient for required safety analyses. Little development will be required to make them applicable to a repository system.

II.F.4 Environmental Considerations

II.F.4.1 Introduction

The long-term disposal of high-level radioactive wastes in deep geologic formations has been under study in the United States since the mid-1950's. Since that time, many geologic and environmental studies have been conducted to provide the basis for the design, construction, and operation of deep underground repositories for radioactive wastes. A broad spectrum of agencies and organizations has sponsored this research, including the Atomic Energy Commission, the Energy Research and Development Administration, the Department of Energy, the Environmental Protection Agency, the Nuclear Regulatory Commission, and the Electric Power Research Institute (538, 548, 549, 789-794). In addition, organizations such as the National Academy of Sciences-National Research Council have evaluated the program and made recommendations (795). Similar studies have been performed in a number of other countries, such as Sweden, West Germany, Canada, Japan, and the United Kingdom, and by international organizations, such as the International Atomic Energy Agency, the Commission of the European Communities, and the Organization for Economic Cooperation and Development/Nuclear Energy Agency (663, 664, 796, 797). In general, those studies have determined that deep geologic disposal operations should result in minimal environmental impact.

II.F.4.2 Requirements

Objective 4 in II.A.1 addresses the environmental acceptability of high-level waste disposal systems. With regard to mined geologic disposal, site selection, design, construction, and operation of mined repositories have been guided by broad criteria developed by the National Academy of Sciences-National Research Council, the International Atomic Energy Agency, Nuclear Regulatory Commission, Environmental Protection Agency, and the Department (191, 536, 798-801). These criteria address such considerations as public health and safety, environmental protection, engineering feasibility, and institutional and socioeconomic impacts. Repositories will be operated under radiation protection standards as promulgated by the Commission and the EPA (534, 802). Nonradioactive effluents will be limited by appropriate EPA and State air and water quality standards (803, 804). Potential environmental impacts from construction, operation, and handling of surplus mined material on local ecologic systems will be evaluated for each candidate site and appropriate mitigation measures proposed. Repository sites will be located to reduce land-use conflicts and appropriate mitigation measures will be adopted following consultation with responsible state and local authorities. Similarly, socioeconomic impacts that may result from repository construction and operation will be addressed.

II.F.4.3 Identification of Impacts

Near-term and long-term environmental impacts have been assessed on a generic and/or site-specific basis, for deep geologic repositories in salt, granite, shale, and basalt (538, 805). As a result of such studies and the guidance and experience accumulated since the implementation of the National Environmental Policy Act, a number of basic factors have been identified which merit consideration in environmental evaluations. Most of the factors listed below are not, of themselves, unique to the mined geologic disposal of radioactive wastes, but may be considered typical of many relatively large-scale industrial activities. These are:

1. Land use, including withdrawal from current usage and conflicts with lands dedicated to special public uses or projected for such uses.

2. Water use during construction and operation.

3. Materials (resource) commitments during construction and operation.

4. Energy Requirements and supply throughout construction and operation.

5. Nonradiological effluents, including discharges into the air and water during construction and operation; waste rock storage and disposal; and off-site noise levels.

6. Radiation dose and health effects, including exposures to people both on-site and off-site during transportation and handling of radioactive materials.

7. Long-term environmental impact, based on site restoration techniques, aesthetics, ecological conditions, and potential radiological impacts.

8. Transportation of personnel, materials, and wastes.

9. Socioeconomic and institutional impacts, such as population modification, service capacity, and management capability.

Environmental impacts of repository and intermediate storage facility construction and operation were evaluated on a generic basis in Section 3.1.5 of the draft Environmental Impact Statement for Management of Commercially Generated Radioactive Waste (Draft EIS) (306). Table II-16 summarizes the major points of the Draft EIS for factors 1 through 6 above. All factors listed are discussed further in II.F.4.3.1 through II.F.4.3.9.

Since the Draft EIS is a generic rather than a site-specific document, a number of clarifying and bounding assumptions were necessary to provide a representative data base. Nevertheless, the impacts presented in Table II-17 and discussed subsequently are realistically conservative and serve as an effective scoping tool for this Statement. Assumptions used in preparing the Draft EIS, and consequently in Table II-16, include:

1. Summation of impacts for waste treatment, transportation, storage, and disposal (807).

2. A light-water reactor scenario that generates 10,000 GWe/yr through the year 2040 (808).

3. Repository availability in the year 2000 (809).

4. Colocation of storage and disposal facilities (810).

5. Worldwide radiation exposure calculations limited to the noble gases, C-14 and H-3, due to their potential for global dispersion (811).

6. Repository heat load (thermal density) limitations of 125 kW/acre in granite and basalt, 40 kW/acre in salt, and 80 kW/acre in shale (812).

7. Spent fuel as the waste form (807).

For the purposes of Table II-16, it was assumed that the requirements for interim and extended storage facilities and the number of repositories for each of the four media considered were as follows: salt--11 storage facilities/8 repositories; granite--11 storage facilities/8 repositories; shale--11 storage facilities/6 repositories; and basalt--11 storage facilities/3 repositories.

Table II-16. Summary of Environmental Effects From Routine Operations of Repositories Becoming Available Starting in the Year 2000 (spent fuel as waste)

Land use	Salt	Granite	Shale	Basalt
Surface facilities, buildings parking lots, hectares (ha)	2×10^3	1×10^3	2×10^3	1×10^3
Access roads, railroads, etc. (ha)	2×10^2	2×10^2	2×10^2	2×10^2
Total property-restricted area (ha)	1×10^4	7×10^3	9×10^3	7×10^3
Additional land on which only subsurface activities will be restricted (ha)	3×10^4	1×10^4	2×10^4	1×10^4

Table II-16 (Continued). Summary of Environmental Effects From Routine Operations of Repositories Becoming Available Starting in the Year 2000 (spent fuel as waste)

	Salt	Granite	Shale	Basalt
Water use				
Construction, m^3 (about 10% is "consumed" in concrete)	3×10^6	3×10^6	3×10^6	3×10^1
Operations, m^3	6×10^7	6×10^7	6×10^7	6×10^7
Materials				
Concrete, m^3	1×10^6	1×10^6	1×10^6	1×10^6
Steel, metric tonne (MT)	8×10^5	1×10^6	8×10^5	1×10^6
Copper, MT	2×10^3	2×10^3	2×10^3	2×10^3
Zinc, MT	1×10^3	1×10^3	1×10^3	9×10^2
Aluminum, MT	3×10^2	4×10^2	4×10^2	3×10^2
Lumber, m^3	5×10^4	5×10^4	4×10^4	4×10^4
Lead, MT	7×10^3	7×10^3	7×10^3	7×10^3
Depleted uranium, MT	3×10^3	3×10^3	3×10^3	3×10^3
Energy				
Coal, MT	1×10^7	8×10^6	1×10^7	8×10^6
Propane, m^3	3×10^4	3×10^4	3×10^4	3×10^4
Diesel fuel, m^3	4×10^6	3×10^6	4×10^6	3×10^6
Gasoline, m^3	2×10^5	2×10^5	2×10^5	2×10^5
Electricity, kWh	2×10^{10}	2×10^{10}	2×10^{10}	2×10^{10}
Man-power, man-yr	3×10^5	1×10^5	3×10^5	2×10^5

Source: (Reference 813) Adapted from Draft DOE/EIS-0046-D Tables 3.1.84, 3.1.85, 3.1.86 and 3.1.87.

Table II-16 (Continued). Summary of Environmental Effects From Routine Operations of Repositories Becoming Available Starting in the Year 2000 (spent fuel as waste)

	Salt	Granite	Shale	Basalt
Nonradiological effluents				
Dust concentration at repository fence, $\mu g/m^3$ reference climate	7×10^1	2×10^2	8×10^1	2×10^2
Dust concentration at repository fence, $\mu g/m^3$ arid climate	8×10^2	2×10^3	9×10^2	2×10^3
Radiation dose commitment for routine operations (70-year total body)				
Regional population, dose due to repository, man-rem (2 million people)	3×10^3	4×10^3	4×10^3	3×10^3
Regional population doses due to naturally occurring sources, man-rem	1×10^7	1×10^7	1×10^7	1×10^7
Worldwide population (6 billion), dose due to a repository, man-rem	2×10^2	2×10^2	2×10^2	2×10^2
Worldwide population dose due to naturally occurring sources, man-rem	4×10^{10}	4×10^{10}	4×10^{10}	4×10^{10}
Work force, man-rem	8×10^4	8×10^4	8×10^4	8×10^4
Health Effects				
Regional population (2 million persons, 70 years), number	0[a]	0	0	0
Worldwide population (6 billion persons, 70 years), number	0	0	0	0

[a] indicates less than one health effect.

Nonradiological accidents and associated injury and fatality rates are discussed in Section 3.1.5 of the Draft EIS. These impacts have been found to be comparable to those experienced in equivalent industries.

Potential environmental impacts of a deep geologic waste repository can be broadly classified as near-term and long-term impacts. Near-term impacts are defined as those that will occur during the operational phase of a repository, including construction. These activities were estimated in the Draft EIS to involve a work force varying in size from an estimated 1,400 to 3,100 persons during construction, and 870 to 2,300 persons during operation depending on the disposal medium (814). Long-term environmental considerations are those that conceivably could occur after a repository is closed. These are primarily concerned with potential long-term releases to the biosphere.

Review of the considerations listed in the table, coupled with the following discussion, indicates that, on a generic basis, repository facility construction and operation will result in minimal environmental impact.

II.F.4.3.1 Land Use

Land use is site-specific. However, in general, the controlled zones around waste management facilities may be several hundred times the area occupied by structures. After decommissioning, the site could be returned to essentially its original condition with the exception of permanent markers as discussed in Section II.E.3.

II.F.4.3.2 Water Use During Construction and Operation

Water use is also site-specific in terms of environmental impact. In areas of abundant water, no significant impact is expected. In the case of the Draft EIS reference environment with an assumed average river flow of 120 m^3/s, for example, water usage would have no significant impact. The 120 m^3/s flow rate is a reasonable bounding assumption for most areas. Potential sites with less access to water resources will require specific evaluations to determine support capacity requirements as was done in the case of the Draft Environmental Impact Statement for the Waste Isolation Pilot Plant (805).

II.F.4.3.3 Materials Commitments for Construction and Operation

Most materials now identified as required for repository construction and operation are not unique and are available in relatively good supply. Stainless steel would represent a special case, insofar as the Draft EIS assessment is concerned; the total requirements are dependent on facility and waste package design. Almost all the nickel and chromium used in the United States is imported, and a more specific estimate of the impact and possible use of alternative materials is required. Assuming that stainless steel requirements do not have a significant impact, the Draft EIS analysis indicates that material commitments do not constitute a deterrent to repository construction and operation. The use of other potentially unique materials for waste package components (II.E.1) will be considered on a case-by-case basis.

II.F.4.3.4 Energy Requirements During Construction and Operation

The predicted energy requirements in Table II-15 appear to be within the range anticipated for most relatively large industrial activities. Actual energy usage must be evaluated on a site-specific basis, since geographic location and facility design will be influencing factors.

II.F.4.3.5 Nonradiological Impacts For Construction and Operation

Nonradiological impacts include such factors as noise, pollutant discharges into the air and water during construction and operation, waste rock storage and disposal, and aesthetics. The use of remote siting could serve to further limit the impact of such factors as excessive or irritating noise levels. Materials from the mined spoils piles could become wind or water-borne during construction and operation. Vehicular and construction generated dusts are also likely. Mitigative measures such as sprinkling for dust control and using sound-absorbing devices for noise control are common and effective corrective measures used in other industries.

II.F.4.3.6 Radiation Dose and Health Effects For Routine Operations

Calculated operational phase population radiation doses from routine disposal operations (assuming all facilities are located in the same region) amounts to less than 0.1% of the dose the same population would receive from naturally occurring sources under the Draft EIS assumptions. Calculated health effects for routine operations are small to nonexistent for waste management and, based on the Draft EIS evaluation, are in accord with Objective 3 in II.A.1.

II.F.4.3.7 Long-Term Environmental Impacts

Long-term impacts of a mined geologic repository are primarily concerned with the potential for radionuclide release to the biosphere. The objectives in II.A.1 and the natural and man-made system criteria in II.D and II.E, respectively, serve to minimize any release. The methods for assessing these impacts were presented in Chapter II.F.1. While the Draft EIS did evaluate long-term releases (815), the systems evaluated were not specifically configured to be consistent with the objectives and criteria in this Statement and currently in use in the NWTS program. Furthermore, the Draft EIS analysis was based on events with probabilities that would have to be considered incredible (conservatively assumed to be 2×10^{-13}/yr probability) (816). More reasonable bounding estimates are presented in the example analyses (Examples 1 and 2) in II.F.1.

Relative to long-term impacts on the ecosystem following repository closure, only one potential effect has been identified. Heat generated by the disposed waste could theoretically result in an increase of ground surface temperature which, being less than $0.5^{o}C$ (less than $1^{o}F$), will hardly affect the species distribution of the native vegetation (817). Heat-induced uplift, if it occurred, might also contribute to slight changes in the near-surface ground water hydrology, and could potentially affect the plant population. Ecological changes resulting from ground heating should occur only over an extended period of time, have impact on only a relatively small area, and be masked by natural variations in the natural succession patterns which normally occur at a relatively high rate (818).

Aesthetic impacts following repository decommissioning will be limited to site identification markers and other intrusion control measures (II.E.3) and perhaps a portion of the rock not used as backfill (819). Return of the site to essentially its original condition will effectively mitigate any long-term visual impact. There will be no noise impacts following decommissioning.

II.F.4.3.8 Transportation Impacts

Transportation requirements and their associated environmental impacts are highly site-specific in that they are related to such factors as distance from established rail and highway systems. Some commitment of land will be required to provide site access and on-site movement capability. Providing new or expanded access to a repository site may result in some alteration of traffic and future growth patterns in the area as well as resulting in air, visual, and noise pollution in the immediate vicinity. An increase in traffic levels can be anticipated due to personnel and materials transport, spoils removal, and the ultimate transport of waste designated for disposal. At least two tangible benefits can be anticipated as a result of the expanded transportation base in the repository region; these are increased attractiveness of the area to new industry and an expanded Federal, State, and local financial base.

The cumulative effects of transportation requirements depend upon existing highway capacity and routing; requirements of the repository; and such mitigative measures as public transportation, vanpooling, and the location of new housing relative to the site. The anticipated volume of materials moving to or from the site and the size of the daily work force are not expected to place excessive demands upon existing transportation systems.

II.F.4.3.9 Socioeconomic and Institutional Impacts

The magnitude of economic impacts depends upon the size and diversity of the economy in the region containing the site. A single repository is not large enough in terms of its work force requirement and expected wage generation to have significant impact on most candidate regions. If multiple facilities are colocated, however, these effects would be increased accordingly. Mitigative actions, such as Federal impact funds, could reduce the severity of adverse impacts. In addition, the economic growth and diversity implied by such a development would normally be viewed as beneficial.

A waste repository is not expected to induce unmanageable levels of growth at most sites. Social impacts are more likely to be of the qualitative sort, reflecting increased social disruption by placing stress on existing social institutions and services. These potential impacts can be effectively mitigated with advanced planning and public participation in decisions relating to the development of the project. Project work force requirements might help to reduce local unemployment (although excess inmigration exceeding the needs of the job could increase unemployment), help to diversify the local economy, and induce growth of secondary economic activities. In addition, the influx of skilled workers might enhance the overall level of education and expertise in local community.

As discussed in Section III.C.2, the Department has undertaken and is continuing an intensive program to interact with the public and with State, local, and Federal governments throughout the repository siting and operation period.

II.F.4.4 Environmental Summary

Results of research and development and environmental assessments indicate that a mined geologic disposal facility for high-level radioactive wastes can be built and operated safely with minimal effects on man and his environment. Land use and water use are site-specific, but the amounts required are small in terms of environmental impact. A repository will pose nonradiological impacts similar to those encountered in a sizable deep-mine type of complex. Local natural and human resources will be drawn upon to the extent possible to construct and operate the new facility. The extent of socioeconomic effects on surrounding communities will depend mainly upon existing conditions in the region in which a repository is located. However, careful planning and appropriate compensatory actions will minimize these impacts.

The potential radiation exposure to the surrounding regional population is only a small percentage of the dose the population would receive from naturally occurring sources and is in accord with Objective 2 in Section II.A.1. After the facility has served its useful life and is decommissioned, it should be possible to return the land surface to natural vegetative cover and normal use, with the exception of those limitations imposed by the requirements for restriction of subsurface activities and the need for permanent markers.

II.G SUMMARY OF TECHNICAL BASIS FOR CONFIDENCE THAT SPENT FUEL WILL BE DISPOSED OF IN A SAFE AND ENVIRONMENTALLY ACCEPTABLE MANNER

Part II presents the technical basis for finding that spent nuclear fuel can be disposed of in a safe and environmentally acceptable manner using technology which can be implemented in a reasonable period of time. The Department has adopted an interim planning strategy which focuses on the disposal of spent fuel in mined geologic repositories in accordance with the program established by the President of the United States on 12 February 1980. The Department's technical programs are (i) sufficiently flexible to meet standards to be promulgated by the Nuclear Regulatory Commission and the Environmental Protection Agency, (ii) sufficiently diverse in scope that high assurance can be provided that acceptable systems will result without undue reliance on the results of any specific R&D effort, and (iii) sufficiently conservative to compensate for the residual uncertainties inherent in mined geologic disposal.

In reaching the above conclusions, the Department has evaluated the existing technical information and the expected results of the research and development program and has assessed the expected performance of mined geologic disposal systems. Key points supporting the Department's conclusions are delineated in paragraphs 1 through 7, which follow:

1. <u>The Department has outlined specific performance objectives for waste disposal systems which reflect the developing consensus from diverse groups as to what must be achieved, and which can serve as a basis for assessing the Department's program until formal NRC and EPA standards are available.</u>

The performance objectives provide program focus and direction pending adoption of regulations and standards now under development (II.A.1). These proposed objectives are based on points of apparent consensus among Federal agencies, scientific groups, informed individuals, and public groups. The objectives are consistent with those expressed by the President in his Message to Congress of 12 February 1980 and by the Interagency Review Group. By focusing on these objectives and by continued participation in rulemaking activities and information exchange meetings with the NRC and the EPA, the Department will ensure that the waste disposal system as finally designed and constructed will meet applicable regulatory requirements. The Department maintains an awareness of the requirements under consideration by those agencies and will maintain the flexibility to meet those requirements they ultimately establish.

2. <u>The Department has adopted a conservative approach for the development of waste disposal systems to ensure that health, safety, and environmental objectives will be met.</u>

The conservative approach adopted by the Department is based upon a step-wise approach to system development and implementation, a multibarrier system for radionuclide containment and isolation, and appropriate design and operating margins to compensate for uncertainties (II.A.2).

Proceeding in a cautious, step-wise manner in the development and implementation of waste disposal systems adds assurance that the best available information is considered in reaching decisions and irreversible impacts are minimized. The use of multiple independent natural and man-made barriers against waste release minimizes the impacts of potential disruptive forces by avoiding undue reliance on any given barrier. The use of appropriate design and operating margins provides assurance that residual uncertainties inherent in disposal systems are compensated for. Integration of scientific peer review into the program adds further assurance that the waste disposal objectives will be met. The Department's approach ensures that the best available pertinent information will be considered in reaching decisions and that a high confidence in safety will be attained in spite of residual uncertainties in data, modeling, or future conditions.

3. <u>Based on an evaluation of several alternate methods for waste disposal, mined geologic disposal has been identified as the focus of the Department's interim waste management planning strategy</u>.

Alternate methods for spent fuel disposal have been compared on the basis of available information and the objectives in II.A.1, and mined geologic disposal has been identified as the most suitable method for implementation in the NWTS Program (Chapter II.B). Mined geologic disposal has been endorsed by a number of recognized scientific review groups and has been selected by other nations for implementation. The IRG recommended and the President recently adopted an interim planning strategy for disposal of high-level waste focused on the use of mined geologic repositories. The Department will continue to investigate other concepts which show promise (primarily sub-seabed and deep hole disposal) as a backup to geologic repositories in order to provide further confidence that the waste disposal objectives will be met.

4. The Department has identified those characteristics and requirements necessary for successful mined geologic disposal.

The characteristics of a successful geologic waste disposal system may be summarized as follows:

4.1 The host and overlying rock will prevent disruption of containment by surficial events.

Wastes will be placed in repositories mined in rock formations several hundred meters beneath the surface of the Earth. The depth of the repository will be chosen on a site-specific basis to avoid losses of containment due to surficial processes, including weathering, erosion, or glaciation, or large, impacting forces (such as meteors) (II.D.2). Deleterious impacts from acts of war or sabotage and acts of man other than those requiring the use of special equipment necessary to reach depths of several hundred meters will also be avoided.

4.2 Interactions between the waste and the ground water systems will be minimized.

The repository will be sited and designed so as to minimize contact between circulating ground water and waste materials in order to minimize the potential for waste transport. The repository system will be located in a geohydrologic system which is highly restrictive to ground water flow into, through, or from the repository structure (II.D.2). Site and host rock characterization will be carried out using state-of-the-art techniques which will provide confidence in the characterization of geologic and hydrologic conditions existing at the site (II.D.4).

The host rock selected will have sufficient stability to retain the desirable containment and isolation properties for which it was chosen, as far as can be reasonably predicted into the future. (II.D.2 and II.F.1). Prediction will be based upon state-of-the-art techniques for predicting the natural forces which could reasonably be anticipated to act on the host rock.

4.3 The repository will be designed to preserve the containment and isolation provided by the natural systems.

The potential impacts to natural systems from the introduction of heat-producing radioactive wastes are undergoing systematic evaluation. Preliminary findings in the areas indicated below provide confidence that a system will be developed to meet the performance objectives.

Thermal effects can be controlled by limiting thermal loadings and thus temperatures. The waste-package environment can thus be controlled preventing prémature degradation of the waste package. Temperatures, stresses, and deformations in the host rock can be limited to achieve safe emplacement and, if necessary, retrieval of wastes (II.E.2 and II.F).

Penetrations into the repository will be sealed to limit the ingress or egress of ground water. Research in progress has established characteristics of shaft and borehole seals in in situ environments. Laboratory programs have been established to assess the long-term stability of candidate sealing materials. Although nearly complete sealing is being sought, shaft and borehole seals would impede significantly the flow of fluids, even if the seals performed at less than design effectiveness (II.E.2).

Potential impacts on the containment and isolation capability of the natural systems due to the process of repository excavation will be avoided by controlling extraction ratios and through the use of appropriate backfill materials and techniques (II.E.2). The large base of technical information from the commercial mining industry, supplemented by the studies described in Chapter II.E, provide a high degree of confidence that excavation-related impacts will be minimized.

4.4 Waste packages are being developed which will ensure containment during the period dominated by fission product decay in all credible repository environments.

The waste package will be designed and constructed to be compatible with host rock characteristics and provide for waste containment. The state of knowledge regarding waste/host rock interactions is such that conservative bounds can be established for allowable interactions (II.C). Components of the waste package are being designed to operate within those bounds (II.E.1). Many candidate materials exhibit the characteristics necessary to satisfy containment requirements. Moreover, knowledge is continuously increasing, thereby providing further confidence in designs which would be used for repositories developed within the schedules proposed (II.E.1).

Waste/ground water contact will be further limited by multibarrier waste packages designed to (i) prevent waste/ground water contact throughout the period dominated by fission product decay and (ii) result in low radionuclide release rates to ground water should waste/ground water contact occur. Based on knowledge gained from the studies performed to date, it is likely that waste packages can be constructed from the materials being evaluated that will accomplish (i) and (ii) (II.F.1). Such packages provide redundant assurance of waste containment beyond that provided by the natural systems.

4.5 The geohydrologic system surrounding the repository will minimize the release of radionuclides if a loss of containment effectiveness occurs.

Notwithstanding the containment and isolation provided by the near-field systems, the repository will be located within a far-field system with geohydrological properties chosen from extensive siting studies for its ability to provide waste isolation (II.D and II.F.1). For example, a site will be chosen where the ground water flow rate is sufficiently slow that ground water travel times between the repository and the biosphere will be on the order of thousands of years, even considering potential long-term changes in climate (II.D.1 and II.F.1).

Many radionuclides will experience significant additional delays due to sorptive processes between the radionuclides and earth minerals. For such nuclides, sorption will result in travel times many times longer than ground water travel times. Radionuclide sorption will increase the overall effective radionuclide travel time from the repository to the biosphere, thereby reducing potential radiological impacts (II.D.2).

4.6 Mined geologic disposal systems can be implemented with technology currently or soon to be available.

Development of a stable underground facility can be achieved in all media under consideration for repositories, using technologies that are presently and/or soon to be available (II.E.2). Moreover, mined geologic disposal provides for retrieval of wastes without excessive complexity. Mined geologic disposal allows for the use of various components of a multiple barrier system. Its development is compatible with the step-by-step strategy (II.A.2). Design and operating margins can be readily incorporated into the system to compensate for residual uncertainties.

4.7 The likelihood and consequences of potential inadvertent human intrusion can be limited by appropriate siting and protective measures.

The waste disposal system will be sited and designed to protect against inadvertent human induced releases, using a multilevel system of protective measures (II.E.3). Such a system will reduce the likelihood of inadvertent human-induced releases to the extent that it is reasonable to do so. Beyond that, the potential consequences of human-induced releases will be reduced to appropriately low levels by the natural and man-made systems described throughout this statement and referred to above.

5. Understanding of these characteristics has been used to develop criteria for host rock and site qualification, establish conceptual designs, and identify elements which warrant more research and development emphasis.

These criteria have allowed the Department to identify areas (as summarized in II.D.3), which have a high likelihood of containing qualified sites. Exploration programs are in progress and have in some cases already resulted in detailed characterization of surface and subsurface environments. Hydrologic data and calculations for sites in granite, basalt, salt domes, and bedded salt have indicated ground water transport times sufficient for waste isolation (Objective 2, II.A.1).

The Department has developed conceptual designs of repositories which include techniques for emplacement and retrieval of spent fuel (II.F.3). These studies have also allowed characterization of the local environment (e.g., temperature) for the waste package and evaluation of operational safety concerns. Using the data from site characterization and conceptual design studies and sensitivity and uncertaintity analyses (II.F.1) programs are under way to reduce the technical uncertainty of sensitive system parameters.

6. <u>System performance assessments, which estimate doses to people resulting from hypothetical disruptive events, indicate that carefully-designed repositories in properly-selected sites will meet the required objectives.</u>

The NWTS program includes development of comprehensive consequence assessment methods. Although still under development and undergoing continual refinement, these techniques have been applied to a large variety of site and repository conditions. Examples of these applications are given in this Statement (II.F.1). Taken together, the studies to date indicate that site characteristics, repository designs, and waste packages can be integrated to achieve acceptable impacts even for events which are highly unlikely but credible. Uncertainties have been compensated for in the models by bounding the reasonable ranges of sensitive parameters to establish the limits of potential impacts. The conclusions from these studies support implementation and further development of mined geologic disposal.

7. <u>The overall objective of the effective isolation of radionuclides from the biosphere in a safe and environmentally acceptable manner will be met.</u>

The mined geologic disposal system will include features which will meet the performance objectives as follows:

7.1 Containment will be virtually complete during the period dominated by fission product decay.

7.2 Isolation will be effective for at least 10,000 years and reasonably foreseeable events will not produce consequences greater than normal variations in background radiation.

7.3 The operational phase of the waste disposal system will be as safe as the operation of other nuclear fuel cycle facilities.

7.4 There are no unreasonable environmental impacts.

7.5 Conservative design and evaluation will compensate for any residual uncertainties.

7.6 Acceptable performance is based on a reasonably available level of technology and is not dependent upon continued maintenance or surveilance for unreasonable times into the future.

7.7 Implementation is independent of the size of the nuclear industry and of the resolution of fuel-cycle or reactor design issues and is compatible with national policies.

References For Part II

(1) Presidential Message to Congress, February 12, 1980, "Comprehensive Radioactive Waste Management Program," Weekly Compilation of Presidential Documents, Vol. 16, No. 7

(2) Interagency Review Group (IRG), Report to the President by the Interagency Review Group on Nuclear Waste Management, NTIS Report TID 29442, Washington, DC, March 1979

(3) Interagency Review Group on Nuclear Waste Management, Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste, TID-28818 (Draft), October 1978

(4) U.S. Nuclear Regulatory Commission, "Means for Improving State Participation in the Siting, Licensing and Development of Federal Nuclear Waste Facilities," NUREG 0539, March 1979

(5) U.S. Environmental Protection Agency, "Proposed EPA Criteria for Radioactive Wastes", 43 Fed. Reg., Sec. 221 (November 1978)

(6) U.S. Nuclear Regulatory Commission, "Draft rule 10 CFR 60," 44 Fed. Reg. 70408 (December 1979)

(7) T.B. Cochran and D. Rotow, Radioactive Waste Management Criteria, Natural Resources Defense Council, Washington, DC, January 1979

(8) National Academy of Sciences, Forum on Radioactive Waste Management, Washington, DC, November 1979

(9) Interagency Review Group on Nuclear Waste Management, Report to the President, TID-29442, p. 46, March

(10) T. B. Cochran, Radioactive Waste Disposal Criteria, Natural Resources Defense Council, Washington, DC, May 1978

(11) J.J. Cohen and K.A. Tonnessen, Survey of Naturally Occurring Hazardous Materials in Deep Geologic Formations: A Perspective on the Relative Hazard of Deep Burial of Nuclear Waste, UCRL-52199, Lawrence Livermore National Laboratory, Livermore, CA, January 1977

(12) U.S. Environmental Protection Agency, Consideration of Environmental Protection Criteria for Radioactive Waste, Background Report, February 1978

(13) Nuclear Energy Agency, Organization for Economic Cooperation and Development, Objectives, Concepts and Strategies for the Management of Radioactive Waste Arising from Nuclear Power Programs, A Report by an NEA Group of Experts, September 1977

(14) T. Greenwood, "Radioactive Waste Management in the United States", p. 28, A paper prepared for the Aspen Institute Conference on Resolving the Social Political and Institutional Conflicts Over the Permanent Siting of Radioactive Waste, Cambridge, MA, November 15-17, 1979

(15) Interagency Review Group (IRT), Report to the President by the Interagency Review Group on Nuclear Waste Management, p. 2, NTIS Report TID-29442, Washington, DC, March 1979

(16) Ibid., Appendix C

(17) Interagency Review Group on Nuclear Waste Management, Report to the President, TID-28817 (Draft), Washington, DC, October 1978

(18) T. Greenwood, "Radioactive Waste Management in the United States," p. 29, A paper prepared for the Aspen Institute Conference on Resolving the Social Political and Institutional Conflicts Over the Permanent Siting of Radioactive Waste, Cambridge, MA, November 15-17, 1979

(19) Interagency Review Group (IRT), Report to the President by the Interagency Review Group on Nuclear Waste Management, p. 15-18, NTIS Report TID-29442, Washington, DC, March 1979

(20) R.A. Kisner, J.R. Marshall, D.W. Turner, J.E. Vath, Nuclear Waste Projections and Source Term Data for 1977, pp. 41-42, Y/OWI/TM-34, Oak Ridge National Laboratory, Oak Ridge, TN, April 1978

(21) S.N. Storch and B.E. Prince, Assumptions and Ground Rules Used in Nuclear Waste Projections and Source Term Data, ONWI-24, Oak Ridge National Laboratory, Oak Ridge, TN, September 1979

(22) U.S. Environmental Protection Agency, Proceedings, "A Workshop on Policy and Technical Issues Pertinent to the Development of Environmental Protection Criteria for Radioactive Wastes," EPA Proceedings (Two Workshops): ORP/CSD-77-1, Reston, VA, 1977; ORP/CSD-77-2, Albuquerque, NM, 1977; ORP/CSD-78-2, Denver, CO, 1978

(23) J.W. Voss, Safety Indices and Their Application to Nuclear Waste Management Safety Assessment, p. 53, PNL-2727, Pacific Northwest Laboratory Richland, WA, April 1979

(24) B.L. Cohen, "High-Level Radioactive Waste from Light-Water Reactors, Review of Modern Physics, 49 (1), January 1977

(25) J.W. Voss, Hazard Measure Comparison of Spent Nuclear Fuel and Uranium Ore, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, April 1980

(26) U.S. Nuclear Regulatory Commission Response to the "Jeannine Honicker Petition for Emergency and Remedial Action: An Overview Regarding Radiation Exposure As Related to the Nuclear Fuel Cycle," 1978

(27) M. Eisenbud, Environmental Radioactivity, p. 197, Academic Press, New York, NY, 1973

(28) U.S. Environmental Protection Agency, Estimates of Ionizing Radiation Doses in the United States, ORP/CSD 72 1, pp. 7-15, 1972

(29) U.S. Nuclear Regulatory Commission, High Level and Transuranic Waste Branch Technical Position-1, 1977

(30) 10 CFR 50

(31) 40 CFR 190

(32) 10 CFR 70

(33) 10 CFR 20

(34) 10 CFR 40

(35) 10 CFR 30

(36) National Environmental Policy Act of 1969, 42 U.S.C. 4321-4361

(37) 40 CFR 1900-1908

(38) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D, April 1979

(39) K.J. Schneider and A.M. Platt, eds., High-Level Radioactive Waste Management Alternatives, BNWL-1900, Battelle Northwest Laboratories, Richland, WA, May 1974

(40) Battelle Pacific Northwest Laboratory, Alternatives for Managing Wastes from Reactors and Post-Fission Operations into LWR Fuel Cycle, ERDA-76-43, Volumes 1-5, Washington, DC, May 1976

(41) U.S. Environmental Protection Agency, Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal, ORP/CSD 79-1, May 1979

(42) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, p. 1-11, DOE/EIS-0046-D, April 1979

(43) Interagency Review Group on Nuclear Waste Management, Report to the President by the Interagency Review Group on Nuclear Waste Management, p. 35, TID-29442, March 1979

(44) U.S. Environmental Protection Agency, Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal, p. 2-19, ORP/CSD 79-1, May 1979

(45) Division of Earth Sciences, Committee on Waste Disposal, The Disposal of Radioactive Waste on Land, NAS-NRC Publication 519, National Academy of Sciences, Washington, DC, 1957

(46) Committee on Radioactive Waste Disposal, Geologic Criteria for Repositories for High-Level Radioactive Wastes, National Academy of Sciences, Washington, DC, August 1978

(47) Office of Nuclear Waste Isolation, National Program for Management of Commercial High-Level Nuclear Waste: Report to the Congress of the United States on Candidate Repository Evaluations, Battelle Memorial Institute, Columbus, OH, January 1980

(48) U.S. Nuclear Regulatory Commission, 10 CFR, Parts 1, 19, 20, 21, 30, 40, 51, 60, and 70, "Disposal of High-Level Radioactive Wastes in Geologic Repositories: Proposed Licensing Procedures;" 44 Fed. Reg., pp. 70408-70421 (December 1979)

(49) Interagency Review Group on Nuclear Waste Management, Report to the President by the Interagency Review Group on Nuclear Waste Management, p. 37, TID-29442, Washington, DC, March 1979

(50) Ibid., p. 50

(51) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 1.12, 1.23, DOE/EIS-0046-D, April 1979

(52) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 3.6.1-3.6.33, DOE/EIS-0046-D, April 1979

(53) K.J. Schneider and A.M. Platt, eds., High-Level Radioactive Waste Management Alternatives, Sec. 6, Volume 3, BNWL-1900, Battelle Northwest Laboratories, Richland, WA, May 1974

(54) U.S. Environmental Protection Agency, Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal, ORP/CSD 79-1, pp. 7-1 to 7-69, May 1979

(55) Battelle Pacific Northwest Laboratory, Alternatives for Managing Wastes from Reactors and Post-Fission Operations into LWR Fuel Cycle, pp. 25.31-25.51, ERDA-76-43, Washington, DC, May 1976

(56) D.R. Horn, B. Horn, and M.N. Delach, "World Wide Distribution and Metal Content of Deep Sea Manganese Deposits," Proceedings of the Symposium on Manganese Nodule Deposits in the Pacific, Department of Planning and Economic Development, Hawaii, October 1972

(57) W.N. Hedeman, Jr., Director, Office of Environmental Review (EPA), letter to C.A. Heath, Director, Division of Waste Isolation (DOE), regarding USEPA comments on DOE/EIS-0046-D, September 27, 1979

(58) D.A. Deese, "London Convention of 1972", Sub-Seabed Disposal of Radioactive Wastes: Prevention or Management, WHOI-77-71, Woods Hole Oceanographic Institution, Woods Hole, MA, December 1977

(59) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D, pp. 3.3.1-3.3.37, April 1979

(60) K.J. Schneider and A.M. Platt, eds., High-Level Radioactive Waste Management Alternatives, Volume 2, BNWL-1900, Battelle Northwest Laboratories, Richland, WA, May 1974

(61) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 1.24, 1.25, DOE/EIS-0046-D, April 1979

(62) U.S. Environmental Protection Agency, Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal, p. 9-8, ORP/CSD 79-1, May 1979

(63) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D, pp. 3.4.1-3.4.22, April 1979

(64) Ibid., p. 1.25

(65) Interagency Review Group on Nuclear Waste Management, Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste, Appendix C, "Isolation of Nuclear Wastes in Geologic Formations Followed by In Situ Melting and Their Incorporation in the Host Rock: Status of Scientific and Technological Knowledge," TID-28818(Draft), Washington, DC, October 1978

(66) U.S. Environmental Protection Agency, Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal, Section 9.1.6, ORP/CSD 79-1, May 1979

(67) K.J. Schnieder and A.M. Platt, eds., High-Level Radioactive Waste Management Alternatives, V.1.2, BNWL-1900, Battelle Northwest Laboratories, Richland, WA, May 1974

(68) Battelle Pacific Northwest Laboratory, Alternatives for Managing Wastes from Reactors and Post-Fission Operations into LWR Fuel Cycle, Section 25.1.6, U.S. Energy Research and Development Administration, ERDA-76-43, Washington, DC, May 1976

(69) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 3.5.1 - 3.5.30, DOE/EIS-0046-D, A 1979

(70) Ibid., pp. 1.25, 1.26

(71) Ibid., pp. 3.7.1-3.7

(72) Battelle Pacific Northwest Laboratory, Alternatives for Managing Wastes from Reactors and Post-Fission Operations into LWR Fuel Cycle, pp. 25.52 - 25.61, U.S. Energy Research and Development Administration, ERDA-76-43, Washington, DC, May 1976

(73) K.J. Schneider and A.M. Platt, eds., High-Level Radioactive Waste Management Alternatives, Section 5, Volume 3, BNWL-1900, Battelle Northwest Laboratories, Richland, WA, May 1974

(74) U.S. Environmental Protection Agency, Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal, pp. 8-1 to 8-27, ORP/CSD 79-1, May 1979

(75) E.J. Zeller, D.L. Saunders, and E.E. Angino, "Putting Radioactive Waste on Ice--A Proposal for an International Radionuclide Depository in Antarctica", Bull. At. Scientists, pp. 4-9 and 50-52, January 1973

(76) C. Bull, "Radioactive Waste Disposal," Science, Vol. 189, pp. 595 et seq., August 1975

(77) Sir Brien Flowers, Chairman, Sixth Report - Nuclear Power and the Environment, Royal Commission on Environmental Pollution, London, September 1976

(78) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 3.8.1 - 3.8.40, DOE/EIS-0046-D, April 1979

(79) Ibid., p. 1-28

(80) O.L. Kedrovskiy et al., Principles Governing the Evaluation of the Safety of Underground Burial of Liquid Radioactive Wastes in Deep Geological Formations, and Ways of Improving Safety, IAEA-SM-243/110, presented at the International Symposium on the Underground Disposal of Radioactive Wastes, held in Otanieni, Finland, July 1979

(81) V.I. Spitsyn and V.D. Balukova, Physical and Chemical Processes Occurring as a Result of the Disposal of Liquid Radioactive Wastes in Deep Formation, IAEA-SM-243/113, presented at the International Symposium on the Underground Disposal of Radioactive Wastes, held in Otanieni, Finland, July 1979

(82) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, p. 3.8.I, DOE/EIS-0046-D, April 1979

(83) W. Delaguna et al., Engineering Development of Hydraulic Leaching as a Method of Permanent Disposal of Radioactive Wastes, ORNL-4359, Oak Ridge National Laboratory, Oak Ridge, TN, 1968

(84) U.S. Energy Research and Development Administration, Final Environmental Impact Statement: Management of Intermediate Level Radioactive Waste, Oak Ridge National Laboratory, Oak Ridge, TN, September 1979

(85) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 3.10.1 - 3.10.48, DOE/EIS-0046-D, April 1979

(86) Interagency Review Group on Nuclear Waste Management, Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste, Appendix F, "Isolation of Radioactive Wastes in Space: Status of Scientific and Technologic Knowledge," TID-28818 (Draft), Washington, DC, October 1978

(87) K.J. Schnieder and A.M. Platt, eds., High-Level Radioactive Waste Management Alternatives, Section 8, BNWL-1900, Battelle Northwest Laboratories, Richland, WA, May 1974

(88) Battelle Pacific Northwest Laboratory, Alternatives for Managing Wastes from Reactors and Post-Fission Operations into LWR Fuel Cycle, Section 26, Volume 4, ERDA-76-43, U.S. Energy Research and Development Administration, May 1976

(89) U.S. Environmental Protection Agency, Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal, Section 6, ORP/CSD 79-1, May 1979

(90) Ibid., p. 6-28

(91) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 3.9.1 - 3.9.28, DOE/EIS-0046-D, April 1979

(92) Ibid., p. 1.29

(93) Interagency Review Group on Nuclear Waste Management, Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste, TID-28818 (Draft), Appendix E, "Partitioning and Transmitting Long-Lived Radioisotopes with Geologic Disposal of Fission Products: Status of Scientific and Technological Knowledge," Washington, DC, October 1978

(94) K.J. Schnieder and A.M. Platt, eds., High-Level Radioactive Waste Management Alternatives, Sections 7 and 9, BNWL-1900, Battelle Northwest Laboratories, Richland, WA, May 1974

(95) Battelle Pacific Northwest Laboratory, Alternatives for Managing Wastes from Reactors and Post-Fission Operations into LWR Fuel Cycle, Sections 16 and 27, U.S. Energy Research and Development Administration, ERDA-76-43, Washington, DC, May 1976

(96) J.O. Blomeke et al., Actinide Partitioning - Transmutation Program, ORNL/TM-6987, Final Report, Oak Ridge National Laboratory, Oak Ridge, TN, March 1980

(97) H.C. Burkholder, M.O. Cloninger, D.A. Baker, and G. Jansen, "Incentives for Partitioning High-Level Waste," Nuclear Technology, Vol. 31, p. 202, 1976

(98) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Wastes, p. 3.9.8 DOE/EIS-0046-D, April 1979

(99) T.H. Pigford and J. Choi, Transmutation of Radionuclides in Power Reactors, UCB-NE 3241, University of California-Berkeley, Berkeley, CA, December 1976

(100) U.S. Department of Energy, Draft Environmental Impact Statement for the Management of Commercially Generated Radioactive Waste, pp. 3.2.1 - 3.2.23, DOE/EIS-0046-D, April 1979

(101) Ibid., pp. 1.23, 1.24

(102) Interagency Review Group on Nuclear Waste Management, Report to the President by the Interagency Review Group on Nuclear Waste Management, p. 59, TID-29442, Washington, DC, March 1979

(103) Interagency Review Group on Nuclear Waste Management, Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste, App. A., p. 15, TID-28818, October 1978

(104) "Report to the American Physical Society by the Study Group on Nuclear Fuel Cycles and Waste Management," Review of Modern Physics, Vol. 50, No. 1, Part II, pp. 121-128, January 1978

(105) Interagency Review Group on Nuclear Waste Management, Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste, App. A., p. 12, TID-28818, October 1978

(106) Stearns-Roger Engineering Co., Conceptual Design Report - National Waste Terminal Storage Repository for Storing Reprocessed Wastes in a Dome Salt Formation, Stearns-Roger, Denver, CO, January 1979

(107) Kaiser Engineers, Conceptual Design Report - National Waste Terminal Storage Repository for Storing Spent Fuel in a Bedded Salt Formation, Kaiser Engineers, Oakland, CA, January 1979

(108) W.H. Baker and F.D. King, Technical Data Summary, Spent Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant, p. A.3, DPSTD-AFCT-77-7, Savannah River Laboratory, Aiken, SC, August 1977

(109) U.S. Department of Energy, Analytical Methodology and Facility Description - Spent Fuel Policy, DOE/ET-0054, p. 1-4 August 1978

(110) R.A. Kisner, J.R. Marshall, D.W. Turner, J.E. Vath, Nuclear Waste Projections and Source Term Data for 1977, pp. 41-42, Y/OWI/TM-34, Oak Ridge National Labortory, Oak Ridge, TN, April 1978

(111) S.N. Storch and B.E. Prince, Assumptions and Ground Rules Used in Nuclear Waste Projections and Source Term Data, ONWI-24, Battelle Memorial Institute, Columbus, OH, September 1979

(112) A.G. Croff, O.W. Hermann, and C.W. Alexander, Calculated, Two-Dimensional Dose Rates from a PWR Fuel Assembly, ORNL/TM6754, Oak Ridge National Laboratory, Oak Ridge, TN, March 1979

(113) H.C. Claiborne, L.D. Rickertsen and R.F. Graham, Expected Environments in a Nuclear Waste/Spent Fuel Repository in Salt, ORNL/TM-7201, Oak Ridge National Laboratory, Oak Ridge, TN, in draft

(114) Office of Waste Isolation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Vol. 4: "Baseline Rock Properties - Salt," Y/OWI/TM-36/4, Oak Ridge National Laboratory, Oak Ridge, TN, April 1978

(115) J.D. Bredehoeft et al., Geologic Disposal of High-Level Radioactive Wastes - Earth Sciences Perspectives, p. 7, USGS Circular 779, U.S. Geological Survey, 1978

(116) R.L. Bradshaw and W.C. McClain, Project Salt Vault: Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, ORNL-4555, Oak Ridge National Laboratory, Oak Ridge, TN, 1971

(117) G.H. Jenks, Effects of Temperature, Temperature Gradients, Stress, and Irradiation on Migration of Brine Inclusions in a Salt Repository, ORNL-5526, Oak Ridge National Laboratory, Oak Ridge, TN, July 1979

(118) J.J. Hohlfelder, Preliminary Report on the Salt Block II Experiment, SAND79-2226, Sandia National Laboratories, Albuquerque, NM, 1980

(119) J.J. Hohlfelder and G.R. Hadley, Measurements of Water Lost from Heated Geologic Salt, SAND79-0462, Sandia National Laboratories, Albuquerque, NM, 1979

(120) Y.B. Katayama, D.J. Bradley. and C.O. Harvey, WISAP Status Report on LWR Spent Fuel IAEA Leach Tests, PNL-3173, Battelle Northwest Laboratories, Richland, WA, September 1979

(121) R.G. Dosch and A.W. Lynch, Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico, SAND78-0297, Sandia National Laboratories, Albuquerque, NM, 1978

(122) J.D. Osnes and K.B. DeJong, Expected Repository Environments in Granite: Thermal Analysis of the Very-Near-Field Region for Spent Fuel Repository in Granite, Technical Letter Memorandum RSI-0048, RE/SPEC, Rapid City, SD, March 1980

(123) R.G. Charlwood and A.S. Burgess, Immobilized Fuel and Reprocessing Waste Vaults: Design Specification and Scope of Work, AECL Technical Memorandum 1-1, Chalk River, Ontario, 1979

(124) A.S. Burgess and P.O. Sandstrom, Immobilized Fuel and Reprocessing Waste Vaults: Preliminary Design Concepts, AECL Technical Memorandum 2-1, Chalk River, Ontario, 1979

(125) A.S. Burgess and P.O. Sandstrom, Immobilized Fuel and Reprocessing Waste Vaults: Preliminary Design Concepts, AECL Technical Memorandum 5-1, Chalk River, Ontario, 1979

(126) M.A. Mahtab and T.D. Wiles, Reprocessing Waste Vault: Container Near-Field Thermal-Rock Mechanics Analyses,"AECL Technical Memorandum 3-6, Chalk River, Ontario, 1979

(127) G.K. Coates, Expected Repository Environments in Granite: Ground Water Composition in Granite Rock, Technical Letter Memorandum RSI-0047, RE/SPEC, Rapid City, SD, March 1980

(128) R.B. Travis, "Classification of Rocks," Quarterly Report of the Colorado School of Mines, Vol. 50, No. 1, Golden, CO, 1955

(129) J.D. Osnes and K.B. DeJong, Thermal Analysis of Very Near Field Region for Spent Fuel Repository in Basalt, RE/SPEC Technical Letter Memorandum RSI-0049, RE/SPEC, Rapid City, SD, March 1980

(130) R.E. Gephart et al., Hydrologic Studies Within the Columbia Plateau, Washington - An Integration of Current Knowledge, RHO-BWI-ST-5, Rockwell Hanford Operations, Richland, WA, October 1979

(131) Geological Society of America and American Geophysical Institute, Bibliography and Index of Geology, current listings

(132) U.S. Department of Interior, Geological Survey, List of Publications, current listings

(133) P.B. King and H.M. Beikman, "Geological Map of the United States," scale 1:2,500,000, U.S. Geological Survey, 1974

(134) K.D. Kirby, Evaluation of the Regional Repository Concept for Nuclear Waste Disposal: Interim Report, Southern Science Applications Institute, 1979

(135) R.E. Anderson, D.H. Eargle, and B.O. Davis, Geologic and Hydrologic Summary of Salt Domes in Gulf Coast Region of Texas, Louisiana, Mississippi, and Alabama, Open File Report USGS-4339-2, U.S. Geological Survey, 1973

(136) J.D. Martinez, D.H. Kupfer, R.L. Thomas, C.G. Smith, and C.R. Kolb, An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Waste, Institute for Environmental Studies, Louisiana State University, Baton Rouge, LA, 1975

(137) Committee on Radioactive Waste Management, National Research Council, Disposal of Solid Radioactive Wastes in Bedded Salt Deposits, National Academy of Sciences, Washington, DC, 1970

(138) J.R. Connolly and L.A. Woodward, Preliminary Inventory of Pre-Cenozoic Clay Shales and Argillites of the Continental United States, SAND79-2015, Sandia National Laboratories, Albuquerque, NM, 1980

(139) G.V. Smith, T.S. Pink, J.R. Lawrence, L.A. Woodward, Klaus Kyle, and A.R. Lappin, Preliminary Survey of Tuff Distribution in Esmeralda, Nye, and Lincoln Counties, Nevada, SAND79-1539, Sandia National Laboratories, Albuquerque, NM, 1980

(140) R.W. Spengler F. Maldonado, J.E. Weir, W.F. Hanna, and G.L. Dixon, Inventory of Granite Masses in the State of Nevada, Open File Report 79-235, U.S. Geological Survey, Denver, CO, 1979

(141) R. Winar and T. Gates, Crystalline Intrusives in the United States and Regional Geologic Characteristics Important for Storage of Radioactive Waste, ONWI-50, Battelle Memorial Institute, Columbus, OH, December 1979

(142) Akers American, Inc., Review of Potential Host Rocks for Radioactive Waste Disposal in the Southeast United States - Southern Piedmont Subregion, prepared for Savannah River Laboratory, Aiken, SC, June 1978

(143) B.R. Erdal, R.D. Aquilar, B.P. Bayhurst, W.R. Daniels, C.J. Duffy, F.O. Lawrence, S. Maestas, P.Q. Oliver, and K. Wolfsberg, Sorption-Desorption Studies on Granite, I. Initial Studies of Strontium, Technetium, Cesium, Barium, Cerium, Europium, Uranium, Plutonium, and Americium, LA-7456-MS, Los Alamos National Scientific Laboratory, Los Alamos, NM, February 1979

(144) K. Wolfsberg, B.P. Bayhurst, B.M. Crowe, W.R. Daniels, B.R. Erdal, F.O. Lawrence, A.E. Norris, and J.R. Smyth, Sorption-Desorption Studies on Tuff, I. Initial Studies with Samples from J-13 Drill Site, Jackass Flats, Nevada, p. 56, LA-7480-MS, Los Alamos National Scientific Laboratory, Los Alamos, NM, April 1979

(145) P. Fritz, J.F. Barker, and J.E. Gale, Groundwater and Isotope Hydrology of Groundwaters in the Stripa Granite: Results and Preliminary Interpretation, LBL-8285 and SAC-12, Lawrence Berkeley Laboratory, Berkeley, CA, 1979

(146) R.G. Dosch and A.W. Lynch, Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico, SAND78-0297, Sandia National Laboratories, Albuquerque, NM, June 1978

(147) A. Brandstetter, M.A. Harwell, G.L. Benson, D.J. Bradley, J.R. Raymond and R.J. Serne, principal investigators, "Waste Isolation Safety Assessment Program," Battelle Northwest Laboratories, Richland, WA, in Technical Progress Report for the Quarter July 1-September 30, 1979, WBS.1.1.b, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1979

(148) E.W. Vine, R.D. Aquilar, B.P. Bayhurst, W.R. Daniels, S.J. Devilliers, B.R. Erdal, F.O. Lawrence, S. Maestas, P.Q. Oliver, J.L. Thompson, and K. Wolfsberg, Sorption-Desorption Studies on Tuff, II. A Continuation of Studies with Samples from Jackass Flats, Nevada, and Initial Studies with Samples from Yucca Mountain, Nevada, LA-8110-MS, Los Alamos National Scientific Laboratory, Los Alamos, NM, 1980

(149) B.R. Erdal, R.D. Aquilar, B.P. Bayhurst, P.Q. Oliver, and K. Wolfsberg, Sorption-Desorption Studies on Argillite, I. Initial Studies of Strontium, Technetium, Cesium, Barium, Cerium, and Europium, LA-7455-MS, Los Alamos National Scientific Laboratory, Los Alamos, NM, March 1979

(150) H. Stephens and S. Sinnock, Thermosphysical Properties of Rocks: A Perspective on Data Needs, Sources, and Accuracy, SAND79-1259C, Sandia National Laboratories, Albuquerque, NM

(151) J.C. Jaeger and N.G.W. Cook, Fundamentals of Rock Mechanics, 3rd ed., John Wiley and Sons, Inc., New York, NY, 1979

(152) D.E. Munson, Comparison of Creep Models for Rock Salt with Mine Measurement, Sandia National Laboratories, Albuquerque, NM, 1980

(153) N.G.W. Cook and P. Witherspoon, Mechanical and Thermal Design Considerations for Radioactive Waste Repositories in Hard Rock, LBL-7073 and SAC-10, Lawrence Berkeley Laboratory, Berkeley, CA, 1979

(154) G.H. Llewellyn, Prediction of Temperature Increases in a Salt Repository Expected From the Storage of Spent Fuel or High-Level Waste, ORNL-ENG/TM-7, Oak Ridge National Laboratory, Oak Ridge, TN, 1978

(155) D.F. McVey, A.R. Lappin, and R.K. Thomas, Test Results and Supporting Analysis of a Near-Surface Heater Experiment in the Eleana Argillite, SAND79-1746C, Sandia National Laboratories, Albuquerque, NM, 1980

(156) F.D. Hansen, Case History Rock Mechanics Examination of the Jefferson Island Salt Mine: II. Laboratory Evaluation of Strength and Creep Deformation Characteristics of Dome Salt Under Confining Pressure, Technical Memorandum Report RSI-0057 (Y/OWI/SUB-77/23303/5), Oak Ridge National Laboratory, Oak Ridge, TN, 1978

(157) A.R. Lappin and W.A. Olsson, Material Properties of Eleana Argillite--Extrapolation to Other Argillaceous Rocks, and Implications for Waste Management, SAND79-1564C, Sandia National Laboratories, Albuquerque, NM, 1980

(158) T. Chan and N.G.W. Cook, Calculation of Thermally Induced Displacements and Stresses for Experiments at Stripa, LBL-7061 and SAC-22, Lawrence Berkeley Laboratory, Berkeley, CA, March 1980

(159) M. Hood, Some Results from Field Investigations of Thermo-Mechanical Loading of a Rock Mass when Heater Canisters are Emplaced in the Rock, Part I, LBL-9392, Lawrence Berkeley Laboratory, Berkeley, CA, March 1980

(160) E.W. Spencer, Introduction to the Structure of the Earth, McGraw-Hill Book Co., New York, NY, 1969

(161) Committee on Waste Disposal, Division of Earth Sciences, Disposal of Radioactive Wastes on Land, National Research Council Publication 519, National Academy of Sciences, Washington, DC, 1957

(162) W. Wamwersik, Interim Summary of Sandia Creep Experiments on Rock Salt from WIPP Study Area, SAND79-0115, Sandia National Laboratories, Albuquerque, NM, 1979

(163) J.A. Blume and Associates, Nevada Test Site Terminal Waste Storage Program; Subtask 1.3: Facility Hardening Studies, Design Cost Scoping Studies, San Francisco, CA, 1978

(164) G. Swan, The Mechanical Properties of the Stripa Granite, LBL-7074 and SAC-03, Lawrence Berkeley Laboratory, Berkeley, CA, 1978

(165) D.B. Grove, The Use of Galerkin Finite-Element Methods to Solve Mass-Transport Equations, Water Resources Report PB-227 532, U.S. Geological Survey, 1978

(166) D.P. Loucks, A Selected Annotated Bibliography on the Analysis of Water Resource Systems, Vol. 6, OWRT/WRSIC-75-201, W76-01517, Office of Water Research and Technology, Washington, DC, 1975

(167) U.S. Geological Survey, Hydrologic Investigations Atlas HA Series, current listings

(168) G. Dagan, Models of Groundwater Flow in Statistically Homogeneous Porous Formations, Water Resources Research, Vol. 15, No. 1, pp. 47-63, 1979

(169) R.H. Simmons and D.H. Carpenter, Technique for Estimating Magnitude and Frequency of Floods in Delaware, Water Resources Report WRI 78-93, USGS/WRO, U.S. Geological Survey, Towson, MD, 1978

(170) O.C. Hettick, Flood Profiles Along the Cedar River, King County, Washington, Water Resources Report WRI 78-84, USGS/WRO, U.S. Geological Survey, Tacoma, WA, 1978

(171) R.F. Flint, Glacial and Quaternary Geology, John Wiley and Sons, Inc., New York, NY, 1971

(172) G.I. Smith, Climate Variation and Its Effect on Our Land and Water, Circular 776, U.S. Geological Survey, 1978

(173) Anco Engineers, Inc., Seismic Assessment of Underground and Buried Nuclear Power Plants, prepared for the Aero Space Corporation, El Segundo, CA, 1977

(174) L.J. Vortman, Prediction of Ground Motion from Underground Nuclear Weapons Tests as it Relates to Siting of a Nuclear Waste Storage Facility at NTS and Compatibility with the Weapons Test Program, 1979

(175) H.Q. Pratt, H.A. Hustnulia, D.E. Stephenson, Earthquake Damage to Underground Facilities, DP-1513, Savannah River Laboratory, Aiken, SC, 1978

(176) P.R. Stevens, A Review of the Effect of Earthquakes on Underground Mines, Open File Report 77-313, U.S. Geological Survey, April 1977

(177) S.T. Algermissen and D.M. Perkins, A Probabilistic Estimate of Maximum Ground Acceleration in the Contiguous United States, Open File Report 76-416, U.S. Geological Survey, 1976

(178) Office of the Director of Mines, Directory of Nevada Mine Operations Active During 1978, Carson City, NV, 1979

(179) R.F. Legget, Geology and Engineering, 2nd ed., p. 400, McGraw-Hill Book Co., New York, NY, 1962

(180) C.F. Park, Jr., and R.A. MacDiamid, Ore Deposits, 2nd ed., W.H. Freeman, San Francisco, CA, 1970

(181) D.R. Mullineaux, "Preliminary Overview Map of Volcanic Hazards in the 48 Conterminous United States," Scale 1:7,500,000 Miscellaneous Field Studies Map MF-786, U.S. Geological Survey, reprinted 1977

(182) B.M. Crowe, Disruptive Event Analysis: Volcanism, prepared for Waste Isolation Safety Assessment Program, Battelle Pacific Northwest Laboratories, Richland, WA, 1978

(183) B.M. Crowe and W.J. Carr, Pliocene and Quaternary Volcanic Rocks of the Southern Great Basin: Status of Volcanic Risk Assessment for the Nevada Nuclear Waste Storage Investigations, Open File Report, U.S. Geological Survey, Denver, CO, 1980

(184) U.S. Department of Energy, Nevada Nuclear Waste Storage Investigations, FY 1979 Program Plan, Nevada Operations Office, Las Vegas, NV, March 1980

(185) R.W. Fairbridge, "Denudation," The Encyclopedia of Geomorphology, R.W. Fairbridge (ed.), pp. 161-271, John Wiley and Sons, Inc., New York, NY, 1968

(186) S. Sinnock, Geomorphology of the Uncompahgre Plateau, Western Colorado, USA, Ph.D. dissertation, Purdue University, West Lafayette, IN, 1978

(187) T.C. Gustavson, Locating Field Confirmation Study Areas for Isolation of Nuclear Waste in the Texas Panhandle, Mid-year Report, March 1980, pp. 7-17, Texas Bureau of Economic Geology, University of Texas at Austin, Austin, TX

(188) Office of Nuclear Waste Isolation Staff, NWTS Criteria for the Geologic Disposal of Nuclear Wastes: Site Qualification Criteria, ONWI-33(2), Battelle Memorial Institute, Columbus, OH, December 1979

(189) G.D. Brunton and W.C. McClain, Geological Criteria for Radioactive Waste Repositories, Y/OWI/TM-47, Oak Ridge National Laboratory, Oak Ridge, TN, 1977

(190) Panel on Geological Site Criteria, Committee on Radioactive Waste Management, National Research Council, Geological Criteria for Repositories for High Level Radioactive Wastes, National Academy of Science, Washington, DC, 1978

(191) International Atomic Energy Agency, Site Selection Factors for Repositories of Solid High Level and Alpha-Bearing Wastes in Geological Formations, Technical Report Series 177, Vienna, Austria, 1977

(192) D.W. Powers, S.J. Lambert, S.E. Shaffer, L.R. Hill, and W.D. Weart, Geologic Characterization Report for the Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico (2 volumes), SAND78-1596, Sandia National Laboratories, Albuquerque, NM, 1978

(193) C.E. Dutton, Tertiary History of the Grand Canyon Distict, with Atlas, U.S. Geological Survey Monograph 2, 1882

(194) O.C. Marsh, "Bird with Teeth," Third Annual Report of the U.S. Geological Survey, J.W. Powell, Dir., pp. 45-88, 1883

(195) G.K. Gilbert, Lake Bonneville, U.S. Geological Survey Monograph 1, 1890

(196) C.F. Marbut, Physical Features of Missouri, Missouri Geological Survey, Vol. 10, pp. 11-109, 1896

(197) C.W. Hayes, The Southern Appalachians, National Geographic Society Monograph, Vol. 1, No. 10, 1895

(198) Kansas Geological Society, Bibliography and Index of Kansas Geology Through 1974, Bulletin 213, July 1974

(199) American Association of Petroleum Geologists, Annual Drilling Reports, current listings

(200) FUGRO, Inc., Preliminary Safety Analysis Report: Sun Desert Power Plant Units 1 and 2, prepared for San Diego Gas and Electric Co., 1977

(201) FUGRO, Inc., San Joaquin Nuclear Project; Early Site Review Report, prepared for Los Angeles Dept. of Water and Power, 1977

(202) Rochester Gas-Electric Corporation, "Sterling Power Project Nuclear Unit No. 1 Addendum Standardized Nuclear Unit Power Plant System SNUPPS," Preliminary Safety Analysis Report, Vol. 1, Section 2.5, 1976

(203) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Geologic Disposal of Radioactive Waste in a Mined Repository, DOE/TIC-11033 (Draft), April 1980

(204) Interagency Review Group (IRG), Report to the President by the Interagency Review Group on Nuclear Waste Management, NTIS Report TID-29442, Washington, DC, March 1979

(205) R.R. Compton, Manual of Field Geology, John Wiley and Sons, Inc., New York, NY, 1962

(206) C.C. Albritton, Jr., ed., Philosophy of Geohistory: 1875-1970 Benchmark Papers in Geology, Academic Press, New York, NY, 1975

(207) C.T. Harper, ed., Geochronology: Radiometric Dating of Rocks and Minerals, Benchmark Papers in Geology, Academic Press, New York, NY, 1973

(208) O.A. Schaeffer and J. Zahringer, eds., Potassium Argon Dating, Springer, New York, NY, 1966

(209) R.W. Spengler, D.C. Muller, and R.B. Livermore, Preliminary Report on the Geology and Geophysics of Drill Hole Ue25a-1, Yucca Mountain, Nevada Test Site, Open File Report 79-1244, U.S. Geological Survey, Denver, CO, 1979

(210) F. Maldonado, D.C. Muller, and J.N. Morrison, Preliminary Geological and Geophysical Data of Ue25a-3 Exploratory Hole, Nevada Test Site, Nevada, USGS-1543-6, U.S. Geological Survey, Denver, CO, September 1979

(211) Panel on the Implementation Requirements of Environmental Radiation Standards, Committee on Radioactive Waste Management, National Research Council, Implementation of Long-Term Environmental Radiation Standards, National Academy of Sciences, Washington, DC, 1979

(212) J.E. Gale and P.A. Witherspoon, An Approach to the Fracture Hydrology at Stripa--Preliminary Results, LBL-7079 and SAC-15, Lawrence Berkeley Laboratory, Berkeley, CA, May 1979

(213) M. Parsons, "Determination of the Hydrogeological Properties of Fissured Rocks," Proceedings 24th Geologic Congress, Montreal, Sec. III, Hydrogeology, pp. 89-99, 1972

(214) E. Romm and B. Pozinenko, "Investigations of Seepage in Fractured Rocks," Trudy VNIGRI, 1963

(215) D. Lamar and P.M. Merrifield, Application of Skylab and E.R.T.S. Imagery to Fault Tectonics and Earthquake Hazards of Peninsular Ranges, California, U.S. Geological Survey Publication PB-247 896/AS, 1975 (1976)

(216) R.G. Ray, Aerial Photographs in Geologic Interpretation and Mapping, Professional Paper 373, U.S. Geological Survey, 1960

(217) M.M. Thompson et al., Manual of Photogrammetry, 3rd ed., American Society of Photogrammetry, 1966

(218) J.T. Smith, Jr. and A. Anson, Manual of Color Aerial Photography, American Society of Photogrammetry, 1968

(219) U.S. Department of Interior, Application of Thermal Imagery and Aerial Photography to Hydrologic Studies of Karst Terrain in Missouri, WRI-77-16, U.S. Geological Survey, Water Resources Investigations, 1977

(220) J.D. Friedman and D. Frank, Thermal Surveillance of Active Volcanoes Using Landsat-1 Data Collection System. Part 3: Heat Discharge From Mount St. Helens, Washington, Open File Report 77-541, U.S. Geological Survey, 1977

(221) M.D. Krohn et al., Discrimination of Hydrothermally Altered Rocks Along the Battle Mountain-Eureka, Nevada, Mineral Belt Using Landsat Images, Open File Report 78-585, U.S. Geological Survey, 1978

(222) W.N. Melhorn, S. Sinnock, and R.P. Mroczynski, "Application of Mechanic Processed ERTS-1 Data to Land Use Inventories in Arid Western Colorado," Proceedings, 4th Annual Conference in Remote Sensing, University of Arizona, Tucson, AZ, pp. 107-124, 1973

(223) W.N. Melhorn and S. Sinnock, Recognition of Surface Lithologic and Topographic Patterns in Southwest Colorado with ADP Techniques, Symposium on Significant Results obtained from the Earth Resources Technology Satellite-1, NASA-Goddard, Vol. 1, Section A, pp. 473-481, 1973

(224) Office of Arid Lands Studies, Proceedings: Fourth Annual Conference on Remote Sensing in Arid Lands, University of Arizona, Tucson, AZ, November, 1973

(225) M.J. Gnolien, Interpretation of Landsat Images, Oman, Open File Report 77-296, U.S. Geological Survey, 1977

(226) H.Th. Verstappen, Remote Sensing in Geomorphology, Elsevier Scientific Publishing Co., New York, NY, 1977

(227) M.B. Dobrin, Introduction to Geophysical Prospecting, (2nd ed.), McGraw-Hill Book Co., New York, NY, 1960

(228) D.R. Makey, Gravity Survey of the Western Mojave Desert, California, pp. 51-73, Professional Paper 316-D, U.S. Geological Survey, 1960

(229) E.B. Fabiano et al., Magnetic Total Intensity in the United States-Epoch 1975.0, Maps I-912, I-913, and I-915, U.S. Geological Survey, 1976

(230) P. Romig, "Applications of Geophysical Methods in Nuclear Waste Disposal Siting," Attachment to Memo from R. W. Craig and P. Romig to Participants of Geologic Media May and October Conferences, January 17, 1980

(231) G.V. Keller and F.C. Frieschknecht, Electrical Methods in Geophysical Prospecting, Pergamon Press, New York, NY, 1966

(232) S.L. Stockton and A.H. Balch, The Utility of Petroleum Exploration Data in Delineating Structural Features within Salt Anticlines, Open File Report 78-591, United States Geological Survey, 1978

(233) P.R. Vail et al., Seismic Stratigraphy and Global Changes of Sea Level, AAPG Memoir No. 26, 1978

(234) R. Robinson et al., Trip Report, U.S. Delegation to U.S./Federal Republic of Germany Bilateral Workshop on In Situ Testing, Berlin, Germany, October 1979

(235) K.M. Waters, S. P. Palmer, and W.F. Farrell, Fracture Detection in Crystalline Rock Using Ultrasonic Shear Waves, LBL-7051 and SAC-19, Lawrence Berkeley Laboratory, Berkeley, CA

(236) S.N. Davis and R.J.M. DeWiest, Hydrogeology, John Wiley and Sons, Inc., New York, NY, 1966

(237) J. Apps et al., Geohydrological Studies for Nuclear Waste Isolation at the Hanford Reservation, Final Report, LBL-8764, VC-70, Vol. II, Lawrence Berkeley Laboratory, Berkeley, CA, 1979

(238) I.J. Winograd and W. Thordarson, Hydrologic and Hydrochemical Framework, South-Central Great Basin, Nevada-California, with Special Reference to the Nevada Test Site, Professional Paper 712-C, U.S. Geological Survey, 1975

(239) W. Lin, Measuring the Permeability of Eleana Argillite from Area 17, Nevada Test Site, Using the Transient Method, UCRL-52604, Lawrence Livermore National Laboratory, Livermore, CA, 1978

(240) P. Fritz, J.F. Barker, and J.E. Gale, Geochemistry, Origin, and Age of Groundwaters in the Stripa (Sweden) Test Mine, presented at International Symposium on the Scientific Bases for Nuclear Waste Management, Materials Research Society, Boston, MA, November 26-30, 1979

(241) G. Jack, "Chemistry of Some Groundwaters in Igneous Rocks," Nordic Hydrology, Vol. 4, No. 4, 1973

(242) C.E. Barr and J.A. Carter, Uranium Isotope Disequilibrium in Groundwaters of Southeastern New Mexico and Implications Regarding Age Dating of Waters, Manuscript 28 in International Atomic Energy Symposium 228, Muenchen, FRG, 1978

(243) I.W. Marine, "The Use of Naturally Occurring Helium to Estimate Ground Water Velocities for Studies of Geologic Storage of Radioactive Wastes", Water Resources Research, Vol. 15, No. 5, 1979

(244) J.D. Bredehoeft, A.W. England, D.B. Stewart, N.J. Trask, and I.J. Winograd, Geologic Disposal of High-Level Radioactive Wastes - Earth Science Perspectives, Professional Paper 779, U.S. Geological Survey, 1978

(245) P.A. Witherspoon, ed., Summary Review of Workshop on Movement of Fluids in Largely Impermeable Rocks, Office of Waste Isolation Report Y/OWI/SUB-77/14223, Oak Ridge National Laboratory, Oak Ridge, TN, 1977

(246) L. Ballou, "Field Testing: Heater Tests, Permeability Measurements," Waste Isolation Projects: FY 1978, pp. 5-7, L.D. Ramspott (ed.), Lawrence Livermore National Laboratory, Livermore, CA, January 1979

(247) J.K. Johnstone, In-Situ Tuff Water-Migration/Heater Experiment: Experimental Plan. SAND79-1276, Sandia National Laboratories, Albuquerque, NM, 1980

(248) D.D. Gonzales and L.R. Hill, Tracer Tests Performed in the Field for WIPP in Southeast New Mexico, SAND79-0981C, Sandia National Laboratories, Albuquerque, NM, 1979

(249) Lawrence Berkeley Laboratory, "Geotechnical Assessment and Instrumentation Needs for Nuclear Waste Isolation in Crystalline and Argillaceous Rocks," Symposium proceedings, July 16-20, 1978, LBL-7096, Earth Science Division, Lawrence Berkeley Laboratory, Berkeley, CA, 1978

(250) C.L. Hubbs and R.R. Miller, "The Zoological Evidence--Correlation Between Fish Distribution and Hydrographic History in Desert Basins of the Western United States," in The Great Basin, with Emphasis on Glacial and Post Glacial Times, Utah Univ. Bull., Vol. 38, No. 20, pp. 20-186, 1948

(251) G.M. Richmond, W. Mullenders, and M. Covemans, Climatic Implications of Two Pollen Analyses From Newly Recognized Rocks of Latest Pliocene Age in the Washburn Range, Yellowstone National Park, Wyoming, Bulletin 1455, U.S. Geological Survey, 1978

(252) A.R. Lappin, Preliminary Thermal Expansion Screening Data for Tuffs, SAND78-1147, Sandia National Laboratories, Albuquerque, NM, 1980

(253) G. de Marsily, E. Ledoux, A. Barbreau, and J. Margat, "Nuclear Waste Disposal: Can the Geologist Guarantee Isolation?," Science, Vol. 197, No. 4303, pp. 519-527, August 1977

(254) P.A. Witherspoon, C.H. Amick, J.E. Gale, and K. Iwai, Observations of a Potential Size-Effect in Experimental Determination of the Hydraulic Properties of Fractures, LBL-8571 and SAC-17, Lawrence Berkeley Laboratory, Berkeley, CA, May 1979

(255) B. Giletti et al., State of Geological Knowledge Regarding Potential Transport of High Level Radioactive Wastes from Deep Continental Repositories, PB-289 947, U.S. Department of Commerce, 1978

(256) M. Hood, H. Carlsson and P.H. Nelson, The Application of Field Data from Heater Experiments Conducted at Stripa, Sweden For Repository Design, Part II, LBL-9392 and SAC-26, Lawrence Berkeley Laboratory, Berkeley, CA, 1979

(257) J. R. Hearst, Technical Concept for Rock Mechanics Tests, Climax Granite, NTS, p. 4, V.C.I.D.-18099, Lawrence Livermore National Laboratory, Berkeley, CA

(258) Committee on Field Tests, "Suggested Methods for Monitoring Rock Movements Using Borehole Extensometers," International Society for Rock Mechanics Commission on Standardization of Laboratory and Field Tests, Document No. 5, Pergamon Press, London, England, 1978

(259) N.R. Warpinski, R.A. Schmidt, D.A. Northrop, and L.D. Tyler, Hydraulic Fracture Behavior at a Geologic Formation Interface: Pre-Mineback Report, SAND78-1578, Sandia National Laboratories, Albuquerque, NM, 1978

(260) F. Press and R. Siever, The Earth, 2nd. ed., W.H. Freeman, San Francisco, CA, 1978

(261) A.M. Rogers, D.M. Perkins, and F.A. McKeown, "A Preliminary Assessment of the Seismic Hazard of the Nevada Test Site Region," Bull. Seis. Soc. of Amer., Vol. 67, No. 6, pp. 1587-1606, December 1977

(262) P.B. Schnabel and H.B. Seed, "Accelerations in Rock for Earthquakes in the Western United States," Bull. Seis. Soc. of Amer., Vol. 63, pp. 501- 516, 1973

(263) Seismological Society of America, Bulletins, current listings, 1911 to present

(264) C.E. Glass and D.B. Slemmons, State-of-the-Art for Assessing Earthquake Hazards in the United States, Miscellaneous Paper S-73-1, U.S. Army Engineers Waterways Experiment Station, 1978

(265) S.T. Algermissen and D.M. Perkins, A Technique for Seismic Zoning, pp. 1-15, U.S. Department of Commerce Technical Report ERL 267-ESL 30, S.T. Harding, ed., 1973

(266) H.S. Lew, E.V. Leyendecker, and R.D. Dikkers, Engineering Aspects of the 1971 San Fernando Earthquake, Building Science Series 40, U.S. Department of Commerce, Superintendent of Documents, Washington, DC, December 1971

(267) C.H. Dowding, "Seismic Stability of Underground Openings," Proceedings, Rockston Conference, Stockholm, Sweden, September 1977

(268) C.H. Dowding and A. Rozen, "Damage to Rock Tunnels From Earthquake Shaking," Jour. of Geotechnical Eng. Div., Vol. 104, No. G.T 2, Amer. Soc. Civil Engrs., 1978

(269) A. Holmes and L. Holmes, Principles of Physical Geology, Third Edition, Halsted Press, New York, NY, 1978

(270) R.E. Anderson, D.H. Eargle, and R.O. Davis, Geologic and Hydrologic Summary of Salt Domes in Gulf Coast Region of Texas, Louisiana, Mississippi and Alabama, Open-file Report USGS-4339-2, U.S. Geological Survey, 1973

(271) C.W. Kreitler, E.G. Wermund, E.H. Guevara, A. Giles, and G. Fogg, Preliminary Evaluation of Salt Domes in East Texas, NWTS Program, U.S. Department of Energy, 1978

(272) J.D. Martinez, R.L. Thomas, D.H.Kupfer, C.G. Smith, Jr., C.R. Kolb, E.J. Newchurch, R.E. Wilcox, T.A. Manning, Jr., M. Romberg, A.J. Lesis, and R.E. Rovik, An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, Institute for Environmental Studies, Louisiana State University, Baton Rouge, LA, 1976

(273) P.P. Mellen, Final Report, Preliminary Investigation of Mississippi Salt Domes, Report submitted to CWI, UCC Nuclear Division, Y 12-36, Y16508V, Oak Ridge, TN, 1976

(274) Netherland, Sewell and Associates, Geologic Study of the Interior Salt Domes of Northeast Texas Salt Dome Basin to Investigate their Suitability for Possible Storage of Radioactive Waste Material as of May, 1976, p. 55, Report to OWI, UCC Nuclear Division, Oak Ridge, TN, 1976

(275) J.O. Ledbetter, W.R. Kaiser, and E.A. Ripperger, Radioactive Waste Management by Burial in Salt Domes, p. 82, AEC Contract AT 40-1-4639, Engr. Mechanics Research Lab., University of Texas, Austin, TX

(276) G.D. Brunton, R.B. Laughon, and W.D. McClain, Screening Specifications for Gulf Coast Salt Domes, Y/OWI/TM-[illegible]3, Oak Ridge National Laboratory, Oak Ridge, TN, 1978

(277) NUS Corporation, Non-geologic Criteria for Radioactive Waste Repositories, Y/OWI/SUB77/16504/10(Draft), Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

(278) P.F. Patchik, The Suitability of Palestine Dome, Anderson County, Texas, for a High-Level Nuclear Waste Repository, ONWI-74, Battelle Memorial Institute, Columbus, OH, 1980

(279) R. Kehle, Tectonic Framework and History, Gulf of Mexico Region, Kehle, Ralph, and Associates, Austin, TX, 1978

(280) K.S. Johnson and S. Gonzales, Salt Deposits in the U.S. and Regional Characteristics Important for Storage of Radioactive Waste, Y/OWI/SUB-741411, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

(281) R.J. Hite and S.W. Lohman, Geologic Appraisal of Paradox Basin Salt Deposits for Waste Emplacement, Open File Report 4339-6, U.S. Geological Survey, 1973

(282) Woodward-Clyde Consultants, Paradox Basin Geologic Exploration Area Characterization Plan, (Draft in review), Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH

(283) H.W. Merrell and staff of the Utah Geological and Mineral Survey, "Mineral Resource Inventory of the Paradox Salt Basin, Utah and Colorado," Utah Geological and Mineral Survey Report of Investigation No. 143, 1979

(284) Woodward-Clyde Consultants, Regional Geology of the Paradox Basin, ONWI-92 (Draft in review), Battelle Memorial Institute, Columbus, OH

(285) T.C. Gustavson et al., Locating Field Confirmation Study Areas for Isolation of Nuclear Waste in the Texas Panhandle, ONWI/79/E-511-00300-4, Bureau of Economic Geology, University of Texas at Austin, Austin, TX, March, 1980

(286) K.S. Johnson, Evaluation of Permian Salt Deposits in the Texas Panhandle and Western Oklahoma for Underground Storage of Radioactive Waste, Y/OWI/SUB-4494/1, Oak Ridge National Laboratory, Oak Ridge, TN, 1976

(287) Stone and Webster Engineering Corp., Regional Geology of the Salina Basin, Battelle/ONWI/Sub-E512-06001, (two volumes), Battelle Memorial Institute, Columbus, OH, 1978

(288) Stone and Webster Engineering Corp., New York: Ohio Technical Update and Summary, Battelle/ONWI/Sub-E512-00600/12, Battelle Memorial Institute, Columbus, OH

(289) R.C. Edward, Geophysical Surveys in the Pasco Basin, RHO-BWI-79-100, Rockwell Hanford Operations, Richland, WA, 1979

(290) C.W. Myers, S.M. Price, J.A. Caggiano, P.M. Cochran, W.J. Czimer, N.J. Davidson, R.C. Edwards, K.R. Fecht, G.E. Holmes, M.G. Jones, J.R. Kunk, R.D. Landon, R.K. Ledgerwood, J.T. Lillie, P.E. Long, T.H. Mitchell, E.H. Price, S.P. Reidel, and A.M. Tallman, Geologic Studies of the Columbia Plateau: A Status Report, RHO-BWI-ST-4, Rockwell Hanford Operations, Richland, WA, 1979

(291) Rockwell Hydrologic Science Unit, Hanford Site Water Table Map, Drawer #A2-38396, Rev. 8, Rockwell Hanford Operations, Richland, WA, December 1979

(292) R.K. Ledgerwood and R.A. Deju, Hydrogeology of the Uppermost Confined Aquifers Underlying the Hanford Reservation, ARH-SA-253, Atlantic Richfield Hanford Company, Richland, WA, 1976

(293) R.E. Gephart, R.C. Arnett, R.G. Baca, L.S. Leonhart, and F.A. Spane, Hydrologic Studies Within the Columbia Plateau, Washington: An Integration of Current Knowledge, RHO-BWI-ST-5, Rockwell Hanford Operations, Richland, WA, 1979

(294) S.D. Malone, Annual Technical Report on Earthquake Monitoring of the Hanford Region, Eastern Washington, Geophysics Program, University of Washington, Seattle, WA, 1979

(295) J.B. Cotter, memorandum to D.L. Vieth, August 3, 1978, "Candidate Repository Site Suitability Determination, Control Block of Syncline Ridge Eleana Formation"

(296) NNWSI Project Officers, "A Status Report on the Evaluation of Nuclear Waste Repository Siting on the Nevada Test Site," p. 31, attached to November 21, 1979 letter from M.E. Gates to S. Myers, 1979

(297) E.B. Eckel, ed., "Nevada Test Site," Geol. Soc. of Amer. Memoir 110, p. 290, Boulder, CO, 1968

(298) F.M. Byers et al., Volcanic Suites and Related Cauldrons of Timber Mountain - Oasis Valley Cauldera Complex, Southern Nevada, Professional Paper 919, U.S. Geological Survey, 1976

(299) J.R. Smyth, B.M. Crowe, P.M. Halleck, and A.W. Reed, A Preliminary Evaluation of the Radioactive Waste Isolation Potential of the Alluvium-Filled Valleys of the Great Basin, p. 22, LA-7962-MS, Los Alamos National Scientific Laboratory, Los Alamos, NM, August 1979

(300) L.D. Tyler, "Evaluation of Tuff as a Waste Isolation Medium," Proceedings of the Symposium on the State of Waste Disposal Technology and the Social and Political Implications, Roy G. Post (ed.), February 26 - March 1, 1979, pp. 199-215, University of Arizona, College of Engineering, Tucson, Arizona

(301) R.L. Iman, J.M. Davenport, R.K. Wadell, H.P. Stephens, and D.I. Leap, "Sensitivity Analysis of Parameters of a Regional Hydrologic Model Nevada Test Site and Vicinity, Nevada-California," Proceedings of the 1979 Department of Energy Statistical Symposium, October 1979, Gatlinburg, TN (in press)

(302) A. Ryall, D.B. Slemmons, and Geoney, "Seismicity, Tectonism, and Surface Faulting in the Western United States During Historic Time," Bull. Seis. Soc. Amer., Vol. 56, p. 1105-1135, 1966

(303) R.C. Lincoln, attachments to memorandum to R.M. Nelson, "Tectonics Peer Review Meeting," October 15, 1979

(304) Dames and Moore, Review of Potential Host Rocks for Radioactive Waste Disposal in the Southeastern United States Triassic Basin Subregions, Office of Waste Isolation, Oak Ridge, TN (undated)

(305) Ebasco Services, Inc., Review of Potential Host Rock for Radioactivity Waste Disposal in the Southeast United States, Southeastern Coastal Plain Subregion, Savannah River Laboratory, Aiken, SC, 1978

(306) Office of Nuclear Waste Isolation, Waste Package Design, Development, and Test Plan, ONWI Draft, Battelle Memorial Institute, Columbus, OH, 1979

(307) Y.B. Katayama, Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water, BNWL-2057, Battelle Northwest Laboratories, Richland, WA, 1976

(308) Y.B. Katayama, and J.E. Mendel, Leaching of Irradiated LWR Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water, PNL-SA-6416, Pacific Northwest Laboratory, Richland, WA, 1977

(309) Y.B. Katayama et al., WISAP Status Report on LWR Spent Fuel IAEA Leach Tests, PNL-3173 (Draft), Pacific Northwest Laboratory, Richland, WA, September 1979

(310) Y.B. Katayama, Spent LWR Fuel Leach Tests, PNL-2982, Pacific Northwest Laboratory, Richland, WA, April 1979

(311) M.G. Seitz et al., "Leach Migration Experiments to Determine the Mobility of Radionuclides in Geologic Media," Trans., Am. Nucl. Soc., 32, p. 385, 1979

(312) M.G. Seitz et al., Studies of Nuclear Waste Migration in Geologic Media, ANL-79-30, 1979

(313) M.A. Molecke, Memorandum to Distribution, "PNL-Sandia Joint High Level Waste Total System Test," Sandia National Laboratories, Albuquerque, NM, January 1980

(314) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository, DOE/TIC-11033 (Draft), Table 4, April 1980

(315) J.W. Braithwaite and M.A. Molecke, Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation, SAND79-1935J, Sandia National Laboratories, Albuquerque, NM, 1979

(316) Office of Nuclear Waste Isolation, ONWI Quarterly Report, ONWI-9(4), Battelle Memorial Institute, Columbus, OH, 1979

(317) M.G. Seitz et al., Fuel Cycle Programs Progress Report, July-September 1978, ANL-79-6, 1979

(318) Office of Nuclear Waste Isolation, Program Plan for Field Tests in Geologic Formations, ONWI-80(I) (Draft), Battelle Memorial Institute, Columbus, OH, December 1979

(319) U.S. Department of Energy, Waste Isolation Pilot Plant, Draft Environmental Impact Statement, Vols. I and 2, DOE/EIS-0026-D, April 1979

(320) Sheldon Meyers, memorandum to H.E. Roser, N. Stetson, and A.G. Fremling, "ETW Organization for Materials Characterization for Materials Characterization," August 22, 1979

(321) The Disposal and Long-Term Isolation of Spent Nuclear Fuel, ONWI-59 (Draft), Battelle Memorial Institute, Columbus, OH, September 19, 1979

(322) U.S. Department of Energy, Draft Environmental Impact Statement: Management of Commercially Generated Radioactive Waste, Vol. 2, App. J, DOE/EIS-0046-D, April 1979

(323) A.B. Johnson, Jr., Behavior of Spent Fuel in Water Pool Storage, BNWL-2256, Battelle Northwest Laboratories, Richland, WA, 1977

(324) R.L. Fish et al., Spent Fuel Characterization for Commercial Wastes and Spent Fuel Packaging Program, TC-1384, Hanford Engineering Development Laboratory, Hanford, WA, May 1979

(325) Bechtel National Inc., An Assessment of LWR Spent Fuel Disposal Options, ONWI-39, Battelle Memorial Institute, Columbus, OH, February 1980

(326) G.H. Jenks, Effects of Gaseous Radioactive Nuclides on Design and Operation of Repositories for Spent LWR Fuel in Rock Salt, ORNL-5578, Oak Ridge National Laboratory, Oak Ridge, TN, December 1979

(327) G.A. Towns, Spent Fuel Disassembly and Canning Program at the Barnwell Nuclear Fuel Plant, AGNS-35900-1.2-45

(328) U.S. Energy Research and Development Administration, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, Vol. 3, Section 17, ERDA-76-43, Battelle Northwest Laboratories, Richland, WA, May 1976

(329) L. Jardine and M.J. Steindler, Nuclear Technology, 43, p. 273, 1979

(330) Office of Nuclear Waste Isolation, "Fission Product Release," Quarterly ONWI Report, ONWI-9(4), Battelle Memorial Institute, Columbus, OH, 1979

(331) Office of Nuclear Waste Isolation, "WISAP-Task 2," ONWI Quarterly Report, ONWI-9(4), Battelle Memorial Institute, Columbus, OH, 1979

(332) E.A. Bryant et al., "OKLO, an Experiment in Long-Term Geologic Storage," Actinides in the Environment, ACS Symposium Series No. 35, Paper 6, pp. 89-102, American Chemical Society, New York, NY, 1976

(333) G.J. McCarthy et al., "Hydrothermal Reactivity of Simulated Nuclear Waste Forms and Water Catalysed Waste-Rock Interactions, Scientific Basis for Nuclear Waste Management, Vol. 1, p. 329, Plenum Press, New York, NY 1979

(334) G.J. McCarthy et al., "Hydrothermal Stability of Spent Fuel and High-Level Waste Ceramics in the Geological Repository Environment," Proceedings of the International Symposium on Ceramics in Nuclear Waste Management, T.D. Chikalla and J.E. Mendel, eds., April 30 - May 2, 1979

(335) W. White, paper presented at 1979 Materials Research Society Meeting, Boston, November 1979

(336) W.W. Morgan, "The Management of Spent CANDU Fuel," Nuclear Technology, 24, p. 409, 1974

(337) R.E. Westerman et al., Preliminary Conceptual Designs for Advanced Packages for the Geologic Disposal of Spent Fuel, PNL-2990, Pacific Northwest Laboratory, Richland, WA, April 1979

(338) N.J. Magnani and J.W. Braithwaite, Corrosion Resistant Metallic Canisters: An Important Contributer to the Multibarrier Concept for Nuclear Waste Disposal, SAND79-1429C, Sandia National Laboratories, Albuquerque, NM, 1979

(339) E.J. Nowak, The Backfill as an Engineered Barrier for Nuclear Waste Management, SAND79-0990C, Sandia National Laboratory Albuquerque, NM, 1979

(340) I. Neretnieks, "Retardation of Escaping Nuclides from a Final Repository," KBS Tochnisk Rapport 30, Stockholm, Sweden, 1977

(341) W.J. Mecham, Method for Storage of Solid Waste, U.S. Patent 3,983,050 (1976), U.S. Patent Office

(342) J.W. Braithwaite, N.J. Magnoni, and J.W. Munford, Titanium Alloy Corrosion in Nuclear Waste Environments, SAND79-2023E, Sandia National Laboratories, Albuquerque, NM, 1979

(343) D.W. Shannon, Corrosion of Iron-Base Alloys Versus Alternate Materials in Geothermal Brines, PNL-2456, Pacific Northwest Laboratory, Richland, WA, November 1977

(344) D.W. Shannon, Geochemical Engineering Program - Progress Report for the Period Ending December, 1977, PNL-2736, Pacific Northwest Laboratory, Richland, WA, September 1978

(345) KBS Report Handling and Final Storage of Unreprocessed Spent Nuclear Fuel, Vol. II, Technical Chapter 5, 1978

(346) Ahlstron, KBS, in NUREG/CP-0005, p. 283-316, December 1978

(347) E. Mattsson, "Corrosion Resistance of Canisters for Final Disposal of Spent Nuclear Fuel," Scientific Basis for Nuclear Waste Management, Vol. 1, p. 271, Plenum Press, New York, NY, 1979

(348) Kircher and Bowman, eds., Effect of Radiation on Materials Components, p. 42, Reinhold Publ. Corp., New York, NY 1964

(349) J.D. Sundquist, "Preliminary Corrosion Test of a Glass-Ceramic Candidate for a Nuclear Waste Canister," Scientific Basis for Nuclear Waste Management, Vol. 1, p. 289, Plenum Press, New York, NY, 1979

(350) Kircher and Bowman, eds., Effect of Radiation on Materials Components, p. 330, Reinshold Publ. Corp., New York, NY, 1964

(351) J.M. Rusin, M.F. Browning, and G.J. McCarthy, "Development of Multibarrier Nuclear Waste Forms," Scientific Basis for Nuclear Waste Management, Vol. 1, p. 169, Plenum Press, New York, NY, 1979

(352) Kircher and Bowman, eds., Effects of Radiation on Materials and Components, p. 44, Reinhold Publ. Corp., New York, NY, 1964

(353) F.P. Robert, G.H. Jenks, and C.D. Bopp, Radiation Effects in Solidified High-Level Wastes, Part I. Stored Energy, BNWL-1944, Battelle Northwest Laboratories, Richland, WA, January 1976

(354) R.P. Turcotte, Radiation Effects in Solidified High-Level Wastes, Part II. Helium Behavior, BNWL-2051, Battelle Northwest Laboratories, Richland, WA, May 1976

(355) J.E. Mendel et al., Annual Report on the Characterization of High-Level Waste Glasses, BNWL-2252, Battelle Northwest Laboratories, Richland, WA, June 1977

(356) W. A. Ross et al., Annual Report on the Characterization of High-Level Waste Glasses, BNWL-2625, Battelle Northwest Laboratories, Richland, WA, June 1978

(357) W.J. Weber et al., "Radiation Effects in Vitreous and Devitrified Simulated Waste Glass," Proceedings of the International Symposium on Ceramics in Nuclear Waste Management in Cincinnati, OH, April 30 - May 2, 1979, P.O. Chikalla and J.E. Mandel, Eds

(358) H.C. Weed et al., "Leaching Characteristics of Actinides from Simulated Reactor Waste Glass," Scientific Basis for Nuclear Waste Management, Vol. 1, p. 141, Plenum Press, New York, NY, 1979

(359) J.E. Mendel, "High-Level Waste Glass," Nuclear Technology, 32, p. 72, 1977

(360) J.A. Kelley, "Evaluation of Glass as a Matrix for Solidification of Savannah River Plant Waste," Radioactive Studies, USDOE Report DP-1397, Savannah River Laboratory, Aiken, SC

(361) D.J. B adley, Leaching of Fully Radioactive High-Level Waste Glass, PNL-2664, Pacific Northwest Laboratory, Richland, WA, September 1978

(362) D.J. Bradley, C.O. Harvey, and R.P. Turcotte, Leaching of Actinides and Technetium from Simulated Radioactive High-Level Waste Glass, PNL-3152, Pacific Northwest Laboratory, Richland, WA, September 1979

(363) J.R. Westsik, Jr., J.W. Shade, and G.L. McVay, Temperature Dependence of Hydrothermal Reactions with Waste Glasses and Ceramics, PNL-SA-7789, Pacific Northwest Laboratory, Richland, WA, 1979

(364) J.R. Westsik, Jr. and R.P. Turcotte, Hydrothermal Reactions of Nuclear Waste Solids, A Preliminary Study, PNL-2759, Pacific Northwest Laboratory, Richland, WA, 1978

(365) R.C. Ewing and R.F. Haakon, Naturally Occurring Glasses; Analogues for Radioactive Waste Forms, PNL-2776, Pacific Northwest Laboratory, Richland, WA, 1979

(366) E.L. White, "Permeability Measurements on Cementitions Materials for Nuclear Waste Isolation," Scientific Basis for Nuclear Waste Management, Vol. 1, p. 471, Plenum Press, New York, NY, 1979

(367) J.A. Stone, "Studies of Concrete as a Host for Savannah River Plant Radioactive Waste," Scientific Basis for Nuclear Waste Management, Vol. 1, p. 443, Plenum Press, New York, NY, 1979

(368) G. Rudolph and R. Koster, "Immobilization of Strontium and Cesium in Intermediate-Level Liquid Wastes by Solidification in Cements," Scientific Basis for Nuclear Waste Management, Vol. 1, p. 467, Plenum Press, New York, NY, 1979

(369) R.O. Lokken, A Review of Radioactive Waste Immobilization in Concrete, PNL-2654, Pacific Northwest Laboratory, Richland, WA, June 1978

(370) J.G. Moore, E. Newman, and G.C. Rogers, Radioactive Waste Fixation in FUETAP (Formed Under Elevated Temperature and Pressures) Concretes - Experimental Program and Initial Results, ORNL/TM 6375, Oak Ridge National Laboratory, Oak Ridge, TN, 1979

(371) D.M. Roy and G.R. Gouda, "High-Level Radioactive Waste Incorporation into Special Cements," Nuclear Technology 40, p. 214, 1978

(372) Sandia National Laboratories, Summary of Research and Development Activities in Support of Waste Acceptance Criteria for the Waste Isolation Pilot Plant, SAND79-1305, Albuquerque, NM, 1979

(373) S. Fried and A. Friedman, "Kinetic Effects in Migration," Radioactive Waste in Geologic Storage, ACS Symposium Series 100 (1979)

(374) E.J. Nowak, The Backfill Barrier as a Component in a Multibarrier Nuclear Waste Isolation System, SAND79-1109 (Draft), Sandia National Laboratories, Albuquerque, NM, 1980

(375) R.A. Cantrue, "Experimental Trace Element Transport by Fluid Flow at Elevated Temperatures; Ion Exchange on Kaolinite," Trans., Amer. Geophys. Union 59, p. 1225, 1978

(376) U.S. Department of Energy, Draft Environmental Impact Statement: Management of Commercially Generated Radioactive Waste, Appendix Q, DOE/EIS-0046-D, April 1979

(377) Proceedings of the WISAP Task 4 Second Contractor Information Meeting, PNL-SA-7352, Pacific Northwest Laboratory, Richland, WA, October 1978

(378) S. Komarneni and D.M. Roy, "Shale as a Radioactive Waste Repository: The Importance of Vermiculite," J. Inorg. Nucl. Chem., 41, pp. 1793-1796, 1979

(379) Electric Power Research Institute, Nuclear Waste Management Status and Recent Accomplishments, Final Report, p. 5-1, EPRI NP-1087, May 1979

(380) National Research Council, A Review of the Swedish KBS-II Plan for Disposal of Spent Nuclear Fuel, p. xi, National Academy of Sciences, Washington, DC, 1980

(381) R.L. Bradshaw and W.C. McClain, eds., Project Salt Vault, A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, ORNL-4555, Oak Ridge National Laboratory, Oak Ridge, TN, April 1971

(382) D.H. Lester et al., Systems Study on Engineered Barriers, Task 3, SAI 01379-990LJ (ONWI/SUB/79E515-02000-1), September 1979

(383) U.S. Energy Research and Development Administration, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, ERDA-76-43, Battelle Northwest Laboratories, Richland, WA, May 1976

(384) U.S. Department of Energy, Long-Term High-Level Waste Technology Program, DOE/SR-WM-79-3 (Draft), December 1979

(385) J.A. Stone et al., Preliminary Evaluation of Alternative Waste Forms for Immobilization of Savannah River Plant High-Level Waste, DP-1545, Savannah River Laboratory, Aiken, SC, December 1979

(386) Allied Chemical Corporation, Program Plan: Calcined Solids Retrieval and Handling, ACI-224 (Rev. 1), April 1978

(387) W.F. Bonner, Spray Solidification of Nuclear Waste, BNWL-2059, Battelle Northwest Laboratories, Richland, WA, 1976

(388) W.J. Bjorklund, Fluid Bed Calcination Experience with Simulated Commercial High-Level Nuclear Waste, BNWL-2138, Battelle Northwest Laboratories, Richland, WA, 1976

(389) K.M. Lamb and H.S. Cole, Development of a Pellet Waste Form for High-Level ICP Zirconia Wastes, ICP-1185, April 1979

(390) J.M. Rusin et al., Multibarrier Waste Forms: Part I. Development, USDOE Report PNL-2668-1, Battelle Northwest Laboratories, Richland, WA, 1978

(391) R.H. Burns, "Solidification of Low- and Intermediate-Level Wastes," At. Energy Rev. 9, p. 547, 1971

(392) R.D. Lokken, A Review of Radioactive Waste Immobilization in Concrete, PNL-2654, Pacific Northwest Laboratory, Richland, WA, June 1978

(393) D.M. Roy, "Cement Based Nuclear Waste Solidification Forms," Proceedings of the Conference on High-Level Radioactive Solid Waste Forms, December 18-21, 1978, Denver, CO, pp. 479-516, ed. L. Casey, NUREG/CP-0005, April 1979

(394) J.A. Stone, Evaluation of Concrete as Matrix for Solidification of Savannah River Plants Waste, USDOE Report DP-1448, Savannah River Laboratory, Aiken, SC, 1977

(395) D.M. Roy and G.R. Gouda, "Hot Pressed Cement in Radioactive Waste Management," Final Report to Nuclear Waste Technology Program, Subcontract BSA-481, Battelle Northwest Laboratories, Richland, WA, August 14, 1974

(396) D.M. Roy et al., "A Low Temperature Ceramic Radioactive Waste Form," Proceedings of the International Symposium on Ceramics in Nuclear Waste Management, pp. 136-142, Cinncinati, OH, April 30 - May 2, 1979, eds. T.D. Chikalla and J.E. Mendel, American Ceramic Society and USDOE Conference, 1979

(397) D.M. Roy et al., Low Temperature Ceramic Waste Form, Characterization of Supercalcine Based Monazite-Cement Composites, Materials Research Society Annual Meeting, Boston, MA, 1979

(398) J.A. Kelley, "Evaluation of Glass as a Matrix for Solidification of Savannah River Plant Waste," Nonradioactive and Tracer Studies, USDOE Report DP-1382, Savannah River Laboratory, Aiken, SC, 1975

(399) E.J. Wheelwright et al., Technical Summary: Nuclear Waste Vitrification Project, PNL-3038, Pacific Northwest Laboratory, Richland, WA, 1979

(400) W.A. Ross, Development of Glass Formulations Containing High-Level Nuclear Waste, PNL-2481, Pacific Northwest Laboratory, Richland, WA, 1978

(401) H.T. Blair, Vitrification of Nuclear Waste Calcines by In-Can Melting, BNWL-2061, Battelle Northwest Laboratories, Richland, WA, 1978

(402) J.M. Lukas et al., Compatibility of Two Idaho Chemical Processing Plant Glasses with Electric Melting Processes, PNL-2751, Pacific Northwest Laboratory, Richland, WA, 1978

(403) D. Gombert, H.S. Cole, and J.R. Berreth, Vitrification of High-Level ICP Calcined Wastes ICP-1177, Allied Chemical Corporation, INEL, Idaho Falls, ID, 1979

(404) P.B. Macedo et al., Fixation of Radioactive Wastes in Glass, p. XI-15, CONF-780304, 1979

(405) J.H. Simmons et al., "Fixation of Radioactive Waste in High-Silica Glasses," Nature 278, p. 729, 1979

(406) G.J. McCarthy, Letter to the Editor, Nuclear Technology (in press), "Crystalline Ceramics from Defense High-Level Wastes"

(407) G.J. McCarthy, "High-Level Waste Ceramics, Materials Consideration, Process Simulations, and Product Characterization," Nuclear Technology 32, p. 92, 1977

(408) G.J. McCarthy and M.T. Davidson, "Ceramic Nuclear Waste Forms, I., Crystal Chemistry and Phase Formation," Ceramic Bulletin 54, p. 783, 1975

(409) G.J. McCarthy, J.G. Pepin, and D.R. Clarke, "Crystal Chemistry and Compatibility Relations Among the Synthetic Minerals in Current Supercalcine Ceramics," Proceedings of the International Symposium on Ceramics in Nuclear Waste Management, Cinncinnati, OH, April 30 - May 2, 1979, eds. T.D. Chikalla and J.E. Mendel

(410) Rockwell International Energy Systems Group, "Development of Tailored Ceramics for Geologic Storage of Nuclear Wastes," Quarterly Progress Report, Oct. 1 - Dec. 31, 1979, U.S. Department of Energy, SRO, Contract DE-AC09-79ET 41900, February 15, 1980

(411) A.E. Ringwood, "Immobilization of High-Level Nuclear Reactor Wastes in SYNROC," Nature 278, p. 219, 1979

(412) R.L. Schwoebel and J.K. Johnstone, The Sandia Titanate, A Brief Overview: Ceramic and Glass Radioactive Waste Forms, p. 101, ERDA CONF-77-102, January 1977

(413) G.J. McCarthy et al., "Applications of Hot Pressing to Radioactive Waste Processing," American Ceramic Society Bulletin 53(4), p. 370, 1974

(414) Samsel and Berreth, Nuclear Technology 33, p. 68, 1978

(415) A.K. De' et al., Ceramic Bulletin 55, No. 5, p. 500, 1976

(416) L.J. Jardine and M.J. Steindler, A Review of Metal-Matrix Encapsulation of Solidified Radioactive High-Level Waste, ANL-78-19, Argonne National Laboratory, Argonne, IL, 1978

(417) L.J. Jardine and M.J. Steindler, "Metal Encapsulation of Ceramic Nuclear Waste," Scientific Basis for Nuclear Waste Management, Vol. I, p. 181, Plenum Press, New York, NY, 1979

(418) J.M. Rusin et al., Multibarrier Waste Forms: Part II. Characterizations and Evaluation, PNL-2668-2, Pacific Northwest Laboratory, Richland, WA, 1979

(419) W.S. Aaron, T.C. Quinby, and E.H. Kobish, "Development of Cermets for High-Level Radioactive Waste Fixation," International Symposium on Ceramics in Nuclear Waste Management, Cincinnati, OH, April 30 - May 2, 1979, CONF-790420

(420) W.S. Aaron, T.C. Quinby, and E.H. Kobish, "Cermet High-Level Waste Forms: A Progress Report," ORNL/TM 6404, Oak Ridge National Laboratory, Oak Ridge, TN, 1978

(421) J.S. Ritchie, E.A. Dokelh, J.S. Busch, and D.L. Watson, "Description of an NWTS Repository for Spent Fuel in Bedded Salt," Proceedings of the National Waste Terminal Storage Program Information Meeting, ONWI-62, Battelle Memorial Institute, Columbus, OH, October 30, 1979

(422) H. Grams, P.F. Gnirk, and R.J. Pine, Conceptual Design of a Radioactive Waste Isolation Facility in Plutonic Rock, 18th U.S. Symposium on Rock Mechanics, Keystone, CO, 1977

(423) C.M. Koplik, D.L. Pentz, R. Talbot, Information Base for Waste Repository Design, Vol. 6, Excavation Technology, NUREG/CR-0495, U.S. Nuclear Regulatory Commission, March 1979

(424) P.F. Gnirk, W.H. Grams, and T.J. Zeller, Design Aspects of the Alpha Repository; VII. Summary of Results for the Conceptual Facility Layout, Room Stability Analysis, and Equipment Selection, RE/SPEC Project Summary Report RSI-0033/ORNL-SUB-4269-14, Oak Ridge National Laboratory, Oak Ridge, TN, November 1978

(425) Stearns-Roger Engineering Co., National Waste Terminal Storage Repository for Storing Reprocessing Waste in a Dome Salt Formation "Conceptual Description," p. 3-15; Stearns-Roger Eng. Co., Denver, CO, 1979

(426) Panel on the Implementation Requirements of Environmental Radiation Standards, Committee on Radioactive Waste Management, Implementation of Long-Term Environmental Radiation Standards: The Issue of Verification, p. 26, National Academy of Sciences, Washington, DC, 1979

(427) B.-O. Svanholm, P.-A. Persson, and B. Larsson, "Smooth Blasting for Reliable Underground Opendings," Proc. First Internal Symposium on Underground Space and Storage, Rockstore FF, Stockholm, Vol. 3, pp. 573-579, Stockholm, Sweden, 1977

(428) W. Hustrulid, "Crystalline Rock Mining Technology," Technical Progress Report for the Quarter October 1 to December 31, 1979, ONWI-9(5), Battelle Memorial Institute, Columbus, OH, 1979

(429) M.S. Guiffre, C.M. Koplik, R.L. Plume, and R. Talbort, Information Base for Waste Repository Design, Vol. 5, Decommissioning of Underground Facilities, NUREG/CR-0495, U.S. Nuclear Regulatory Commission, March 1979

(430) Panel on the Implementation Requirements of Environmental Radiation Standards, Committee on Radioactive Waste Management, Implementation of Long-Term Environmental Radiation Standards: The Issue of Verification, p. 28, National Academy of Sciences, Washington, DC, 1979

(431) U.S. Department of Energy, Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste. Vol. 1, Sec. 3.1.34, April 1979

(432) H.C. Claiborne, L. D. Rickertsen and R.F. Graham, Expected Environments in a Nuclear Waste/Spent Fuel Repository in Salt, ORNL/TM-7201, Oak Ridge National Laboratory, Oak Ridge, TN, in Draft

(433) Interagency Review Group Report to the President by the Interagency Review Group on Nuclear Waste Management, Appendix A, p. 31, NTIS Report TID-29442, Washington, DC, March 1979

(434) Panel on the Implementation Requirements of Environmental Radiation Standards, Committee on Radioactive Waste Management, Implementation of Long-Term Environmental Radiation Standards: The Issue of Verification, p. 27, National Academy of Sciences/National Research Council, Washington, DC, 1979

(435) J.D. Bredehoeft, A.W. England, D.B. Stewart, N.J. Trask, and I.J. Winograd, Geologic Disposal of High-Level Radioactive Wastes--Earth-Sciences Perspectives, p. 7, USGS Circular 779, U.S. Geological Survey, 1978

(436) Panel on the Implementation Requirements of Environmental Radiation Standards, Committee on Radioactive Waste Management, Implementation of Long-Term Environmental Radiation Standards: The Issue of Verification, p. 19, National Academy of Sciences, Washington, DC, 1979

(437) Panel on Geologic Site Criteria, Committee on Radioactive Waste Management, Geological Criteria for Repositories for High-Level Radioactive Wastes, p. 11, National Academy of Sciences/National Research Council, Washington, DC, August 1978

(438) J.E. Russell, Areal Thermal Loading Recommendations for Nuclear Waste Repositories in Salt, Y/OWI/TM-37, Oak Ridge National Laboratory, Oak Ridge, TN, June 1979

(439) J.L. Ratigan and P.F. Gnirk, Comments on Thermal/Hydrological/Rock Mechanics Constraints for Selection of a Host Rock and a Geological Site for a Radioactive Waste Repository, Tech. Letter Memo. RSI-0027, RE/SPEC, Rapid City, SD, January 24, 1979

(440) R.G. Charlwood, M.A. Mahtab, A.S. Burgess, D.R. McCreath, P.F. Gnirk, and J.L. Ratigan, "Geological Engineering Factors in the Design of a Radioactive Waste Repository in Hard Crystalline Rock," Geol. Survey of Canada Paper 79-10 (1979-a).

(441) R.G. Charlwood, A.S. Burgess, P.D. Sandstrom, M.A. Mahtab, P.F. Gnirk, and J.L. Ratigan, "Design Concepts for Underground Disposal of Irradiated Candu Fuel and Reprocessing Wastes in Crystalline Rocks," Canadian Nuclear Assoc.: 19th Annual Int. Conf., Toronto, Ontario, (1979-b)

(442) M.P. Hardy, C.M. St.John and G. Hocking, Numerical Modeling of Rock Stresses Within a Basaltic Nuclear Waste Repository, Phase I - Problem Definition, RHO-BWI-C-24, Rockwell Hanford Operations, Richland, WA, April 1978

(443) U.E. Lindblom, P.F. Gnirk and R.G. Charlwood: "Groundwater Movements Around a Repository: Final Report," KBS Teknisk Rapport 54:06, Stockholm, Sweden, 1977

(444) Bechtel National, Inc., National Waste Terminal Storage Conceptual Reference Repository Description, Vol. III, ONWI/SUB/79/E512-01600.16, Battelle Memorial Institute, Columbus, OH, September 1979

(445) R.D. Cheverton and W.D. Turner: Thermal Analysis of the National Radioactive Waste Repository: Progress Through March, 1972, ORNL-4789, Oak Ridge National Laboratory, Oak Ridge, TN, September 1972

(446) G.D. Callahan and J.L. Ratigan, Thermoelastic Analysis of Spent Fuel Repositories in Bedded and Dome Salt, RE/SPEC Technical Memorandum Report RSI-0054 (Y/OWI/SUB-77/22303/4), RE/SPEC, Rapid City, SD, March 1978

(447) G.D. Callahan, Inelastic Thermomechanical Analysis of a Generic Bedded Salt Repository, RE/SPEC Topical Report RSI-0087, RE/SPEC, Rapid City, SD, 1979

(448) M.P. Hardy and G. Hocking, Numerical Modeling of Rock Stresses Within a Basaltic Nuclear Waste Repository, Phase II - Parametric Design Studies, RHO-BWI-C-23, Rockwell Hanford Operations, Richland, WA, July 1978

(449) M.P. Hardy and G. Hocking, Numerical Modeling of Rock Stresses Within a Basaltic Nuclear Waste Repository, Final Report, RHO-BWI-C-32, Rockwell Hanford Operations, Richland, WA, October 1978

(450) F. Holzer and R. Ramspott, eds., Proceedings of a Workshop on Thermomechanical Modeling for a Hardrock Waste Repository, UCAR-10043, Lawrence Livermore Laboratory, Livermore, CA and Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, June 1979

(451) P.R. Dawson, Constitutive Models Applied in the Analysis of Creep of Rock Salt, SAND79-0137, Sandia National Laboratories, Albuquerque, NM, 1979

(452) Rockwell Hanford Operations, Basalt Waste Isolation Project Annual Report - Fiscal Year 1979, RHO-BWI-79-100, Richland, WA, November 1979

(453) Office of Waste Isolation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Vol. 5, "Baseline Rock Properties - Granite," Oak Ridge, TN, April 1978

(454) Office of Waste Isolation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Vol. 7, "Baseline Rock Properties - Basalt," Oak Ridge, TN, April 1978

(455) L.L. Van Sambeek, "Dome Salt Thermomechanical Experiments at Avery Island, Louisiana," Proceedings of the NWTS Program Information Meeting, October 1979, ONWI-82, Battelle Memorial Institute, Columbus, OH

(456) U.S. Department of Energy, Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste, Vol. 2, App. K, Washington, DC, April 1979

(457) Office of Waste Isolation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Vols. 4-19, "Thermal Analyses," Oak Ridge, TN, April 1978

(458) Office of Waste Isolation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Vol. 20, "Thermomechanical Stress Analysis and Development of Thermal Loading Guidelines," Oak Ridge, TN, April 1978

(459) J.L. Ratigan, "Groundwater Movements Around a Repository - Rock Mechanics Analyses," KBS Teknisk Rapport 54:04, September 1977

(460) R.A. Wagner, Parametric Study Involving Thermo/Viscoelastic Analyses of a Room and Pillar Configuration, RE/SPEC Technical Memorandum Report RSI-0070 (ONWI/SUB-78/E512-02300/1), RE/SPEC, Rapid City, SD, November 23, 1978

(461) J.D. Osnes, R.A. Wagner, and H. Waldman, Parametric Thermoelastic Analysis of High-Level Waste and Spent Fuel Repositories in Granite and Other Non-Salt Rock Types, RE/SPEC Technical Memorandum Report RSI-0066 (Y/OWI/SUB-78/22303/12), RE/SPEC, Rapid City, SD, October 13, 1978

(462) A.S. Burgess, R.G. Charlwood, E.L. Skiba, J.L. Ratigan, P.F. Gnirk, H. Stille, and U.E. Lindblom, Analyses of Groundwater Flow Around a High-Level Waste Repository in Crystalline Rock, OECD Nuclear Energy Agency Workshop on Low-Flow, Low Permeability Measurements in Largely Impermeable Rocks, Paris, France, March 19-21, 1979

(463) T.R. Anthony and H.E. Cline, "Thermal Migration of Liquid Droplets through Solids," Journal of Applied Physics, 42, p. 3380, 1971

(464) J.J. Hohlfelder and G.R. Hadley, "Letters of Heat and Mass Transfer," Laboratory Studies of Water Transport in Rock Salt, 6 (4), pp. 271-279, 1979

(465) W.S. Fyfe, N.J. Price, and A.B. Thompson, Fluids in the Earth's Crust, Elsevier Science Publication Company, New York, NY, 1978

(466) S.J. Lambert and H.C. Shefelbine, A Strategy for Investigation of Fluid Migration in Evaporites (Waste Isolation Pilot Plant - WIPP), SAND 79-1889, Sandia National Laboratories, Albuquerque, NM

(467) C.R. Roedder, "Fluid Inclusions in Core Samples from ERDA No. 9 Borehole, WIPP Site, New Mexico," Proceedings of 1978 Materials Research Society, Boston, MA, 1978

(468) P.F. Gnirk, A.F. Fossum, and W.B. Krause, State-of-the-Art Review of Brine Migration Studies in Salt, RE/SPEC Technical Memorandum Report RSI-0075, Sandia National Laboratories, Albuquerque, NM, February 1979 (revised January 1980)

(469) G.H. Jenks, Effects of Temperature, Temperature Gradients, Stress and Irradiation on Migration of Brine Inclusions in a Salt Repository, ORNL-5526, Oak Ridge National Laboratory, Oak Ridge, TN, July 1979

(470) S.J. Lambert, Mineralogical Aspects of Fluid Migration in the Salt Block II Experiment, SAND79-2423, Sandia National Laboratories, Albuquerque, NM, 1980, in draft

(471) P.D. Fairchild and G.H. Jenks, Avery Island - Domed Salt in Situ Tests, p. 4, Y/OWI/Tm-55, Oak Ridge National Laboratory, Oak Ridge, TN, June 1978

(472) A.R. Lappin and W.A. Olsson, "Material Properties of Eleana Argillite--Extrapolation to Other Argillaceous Rocks, and Implications for Waste Management," Proceedings of the NEA/OECD Workshop on Utilization of Agrillaceous Materials for Isolation of Radioactive Wastes, Paris, France, September 10-12, 1979, SAND79-1564C, Sandia National Laboratories, Albuquerque, NM, 1979 (not yet published)

(473) L.D. Tyler, "Model Development for In-situ Test Results in Argillaceous Rock," Proceeding of a Workshop on Thermomechanical Modeling for a Hardrock Waste Repository, Lawrence Livermore Laboratory, Livermore, CA, and Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, June 1979

(474) F.J. Turner, Metamorphic Petrology: Mineralogical and Field Aspects, McGraw Hill Book Co., New York, NY, 1968

(475) J.O. Blomeke, E. Sonder, J.P. Nichols, S. Lindenbaum, R.S. Dillon, E.D. Arnald, and H.F. Soard, An Analsyis of Energy Storage and Its Effects in the Proposed National Radioactive Waste Repository, ORNL-TM-3403 Oak Ridge National Laboratory, Oak Ridge, TN, 1971

(476) C.M. Koplik et al., Information Base for Waste Repository Design, Vol. 3 Waste/Rock Interactions, NUREG/CR-0495, U.S. Nuclear Regulatory Commission, March 1979

(477) G.H. Jenks, Radiolysis and Hydrolysis in Salt-Mine Brines, ORNL-TM-3717, Oak Ridge National Laboratory, Oak Ridge, TN, March 1972

(478) J.W. Braithwaite and M.A. Molecke, Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation, SAND79-1935J, Sandia National Laboratories, Albuquerque, NM, October 1979

(479) K.J. Swyler, R.W. Kaffky, and P.W. Levy, "Radiation Damage Studies on Natural and Synthetic Rock Salt for Waste Disposal Applications," Proceedings of Symposium on Science Underlying Radioactive Waste Management, Materials Research Society, Boston, MA, 1978

(480) International Atomic Energy Agency, Site Selection Factors for Repositories of Solid High Level and Alpha Bearing Wastes in Geologic Formations, p. 40, TRS-177, International Atomic Energy Agency, Vienna, Austria, 1977

(481) G.H. Jenks and C.D. Bopp, Storage and Release of Radiation Energy in Salt in Radioactive Waste Repositories, ORNL-5058, Oak Ridge National Laboratories, Oak Ridge, TN, October 1972

(482) V.I. Spitsyn et al., Investigation of Radiation Resistance of Natural Rock Salt, IAFA International Symposium on Underground Disposal of Radioactive Wastes, Oteneimi, Finland, July 2-6, 1979

(483) G.H. Jenks, Gamma Radiation Effects in Geologic Formation of Interest in Geologic Waste Disposal: A Review and Analysis of Available Information and Suggestions for Additional Experimentation, ORNL-TM-4827, Oak Ridge National Laboratory, Oak Ridge, TN, September 1975

(484) N.E. Carter, letter to J.O. Neff dated May 22, 1979; "Salt Test Facility Justification," Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH

(485) J.T. Thomas, letter to P.D. Obrien dated August 18, 1978, Oak Ridge National Laboratory, Oak Ridge, TN

(486) U.S. Department of Energy, Safety Assessment Document for the Spent Fuel Handling and Packaging Program Demonstration at the Nevada Test Site, NVO-198, Nevada Operations Office, Las Vegas, NV

(487) Kaiser Engineers, "Report of Nuclear Engineering Analytical Support," Conceptual Design Calculations and Studies, Book 11 of 14, 77018 NWTS 3358, December 1978

(488) International Atomic Energy Agency, "The Oklo Phenomenon," Proceedings of the Symposium on the Oklo Phenomenon, Libreville, Gabon, June 23-27, 1975

(489) G.A. Cowan, "A Natural Fission Reactor," Scientific American, 235, pp. 36-47, 1976

(490) F.W. Bingham and G.E. Barr, Scenarios for Long-Term Release of Radionuclides from a Nuclear Waste Repository in the Los Medanos Region of New Mexico, SAND78-1730, Sandia National Laboratories, Albuquerque, NM (not yet published)

(491) D.C. Brookins, Geochemical Constraints on Accumulation of Actinide Critical Mass from Stored Nuclear Waste in Natural Rock Repositories, Technical Report ONWI-17, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, January 1979

(492) Office of Nuclear Waste Isolation The Status of Borehole Plugging & Shaft Sealing for Geologic Isolation of Radioactive Waste, ONWI-15, p. III-2, Battelle Memorial Institute, Columbus, OH, January 1979

(493) Office of Nuclear Waste Isolation, Program Plan and Current Efforts in Repository Sealing for the NWTS Program, p. 12, ONWI-54, Battelle Memorial Institute, Columbus, OH, October 1979

(494) Ibid., p. 5

(495) F. Burns, Needed Research and Development in Repository Sealing: Recommendations of the Earth Sciences Technical Plan Sealing Subgroup, p. I, Battelle Memorial Institute, Columbus, OH, November 1979

(496) Office of Nuclear Waste Isolation, The Status of Borehole Plugging & Shaft Sealing for Geologic Isolation of Radioactive Waste, pp. 4, 5, ONWI-15, Battelle Memorial Institute, Columbus, OH, January 1979

(497) F. Burns, Needed Research and Development in Repository Sealing: Recommendations of the Earth Sciences Technical Plan Sealing Subgroup, p. 9, Battelle Memorial Institute, Columbus, OH, November 1979

(498) Ibid., p.7

(499) C.M. Koplik, D.L. Pentz, and R. Talbot, Information Base for Waste Repository Design, Vol. 1, Borehole and Shaft Sealing, NUREG/CR-0495, U.S. Nuclear Regulatory Commission, January 1979

(500) J. Herndon and L.K. Smith, Plugging Wells for Abandonment: A State-Of-The-Art Study and Recommended Procedure, (Y/OWI/SUB-76/99068), Haliburton Services, Duncan, OK, September, 1976

(501) F. Burns, Needed Research and Development in Repository Sealing: Recommendations of the Earth Sciences Technical Plan Sealing Subgroup, pp. 7, 9, 15, Battelle Memorial Institute, Columbus, OH, November 1979

(502) Ibid.

(503) L.H. Eilers, Borehole Sealing Final Report, p. 101, ORNL/SUB-73/15966, Dow Chemical Company, Dowell Division, Tulsa, OK, 1973

(504) D.M. Roy and W.B. White, Borehole Plugging by Hydrothermal Transport, A Feasibility Report, p. 252, ORNL/SUB-4091/1, The Materials Research Laboratory, The Pennsylvania State University, University Park, PA, 1975

(505) D.L. Black et al., A Study of Borehole Plugging in Bedded Salt Domes by Earth Melting Technology, p. 219, WANL-TME-2870, Westinghouse Electric Corporation, Astronuclear Laboratory, Pittsburgh, PA, 1975

(506) Karn-Bransle-Sakerhet (KBS), Handling and Final Storage of Unreprocessed Spent Nuclear Fuel, Vol. 1, pp. 48-49, 61-65, Karn-Branser-Sakerht, Stockholm, Sweden, 1978

(507) R.T. Martin, Feasibility of Sealing Boreholes with Compacted Natural Earthen Material, Vols. I-III, MIT Research Report 75-28 (ORNL-SUB-3960/2), Massachusetts Institute of Technology, Cambridge, MA

(508) Office of Waste Isolation, A Systems Analysis of Instrumentation for In-Situ Examination of Rock Properties, Final Report, Y/OWI/SUB76/99096 and WANL-TME-2877, Oak Ridge National Laboratory, Oak Ridge, TN, 1976

(509) R.H. Snow and D.S. Chang, Prediction of Cavity Growth by Solution of Salt Around Boreholes, IITRI C6313-14, Illinois Institute of Technology Research Institute, Chicago, IL, 1975

(510) T.N. Dixon, Analysis of Solutioning of Salt Around Well Boreholes, Final Report, University of Texas, Austin, TX, 1973

(511) R.F. Walters, Salt Dissolution in Oil and Gas Test Holes in Central Kansas, Walters Drilling Company, Wichita, KA, 1975

(512) Office of Nuclear Waste Isolation, Program Plan and Current Efforts in Repository Sealing for the NWTS Program, ONWI-54, Battelle Memorial Institute, Columbus, OH, October 1979

(513) C.W. Gulick, Sr, Borehole Plugging Program, Plugging of ERDA-10 Drill Hole, SAND79-0789, Sandia National Laboratories, Albuquerque, NM, June 1979

(514) D.M. Roy and W.B. White, Borehole Plugging by Hydrothermal Transport, Final Report, ORNL-Sub-4091-3, Oak Ridge National Laboratory, Oak Ridge, TN, February 28, 1976

(515) D.M. Roy, Borehole Cement and Rock Properties Studies, Annual Progress Report, Y/OWI/SUB-4091/6, Office of Waste Isolation, Oak Ridge, TN, September 30, 1977

(516) D.M. Roy, B.E. Scheetz, F.L. White, and M. Daimon, Borehole Cement and Rock Properties Studies, Part A. Borehole Plugging Annual Progress Report, ONWI-5, Battelle Memorial Institute, Columbus, OH, October 30, 1978

(517) L.H. Eilers, Sealing AEC #1 Well, Lyons, KA, Final Report, p. 45, ORNL/SUB-74/33542, Dow Chemical Company, Dowell Division, Tulsa OK, 1974

(518) D'Appolonia Consulting Engineers, Inc., The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive Waste, ONWI 15, Battelle Memorial Institute, Columbus, OH, January 1979

(519) D.M. Roy et al., "Borehole Cement and Rock Properties Studies: Task I. Borehole Plugging Cement Studies," Annual Progress Report, October 1, 1978 - September 30, 1979, ONWI/SUB/78/E512-00500-4 (Task I), Battelle Memorial Institute, Columbus, OH, November 30, 1979

(520) D.M. Roy, M.W. Grutzeck, and P.H. Licastro, Evaluation of Cement Borehole Plug Longevity, ONWI-30, Battelle Memorial Institute, Columbus, OH, February 1979

(521) D.M. Roy et al., Geochemistry of Cement-Based Borehole Plugging and Shaft Sealing Systems, Interim Progress Report, May 1, 1979 - September 30, 1979, ONWI/SUB/78/E512-04200-2, Battelle Memorial Institute, Columbus, OH, November 15, 1979

(522) C.L. Christensen and T.O. Hunter, Waste Isolation Pilot Plan (WIPP) Borehole Plugging Program Description, January 1, 1979, SAND79-[illegible], Sandia National Laboratories, Albuquerque, NM, 1979

(523) A.B. Buck and J. P. Burkes, Examination of Grout and Rock from Durral Mine, New Mexico, Miscellaneous Report SL-79-16, Structure Laboratory, U.S. Army Engineers Waterway Experiment Statiuon, Vicksburg, MI, July 1979

(524) B.E. Scheetz, M.W. Grutzeck, L.D. Wakeley, and D.M. Roy, Characterization of Samples of a Cement-Borehole Plug in Bedded Evaporites from Southeastern New Mexico, ONWI-70, Battelle Memorial Institute, Columbus, OH, July 1979

(525) C.L. Christensen, Test Plan, Bell Canyon Test, WIPP Experimental Program, Borehole Plugging, SAND79-0739, Sandia National Laboratories, Albuquerque, NM, June 1979,

(526) C.L. Christensen, Borehole Plugging Program, ONWI Quarterly Report #1-80, October 1, 1979 - December 31, 1979, Battelle/ONWI Subcontract #E511-03700, Sandia National Laboratories, Albuquerque, NM, January 1980

(527) C.M. Koplik et al., Information Base for Waste Repository Design, NUREG/CR-0495, U.S. Nuclear Regulatory Commission, March 1979

(528) J.L. Ratigan and R.A. Wagner, Thermomechanical Analysis of Crushed-Salt Backfilled Disposal Rooms in a Conceptual Radioactive Waste Repository in Dome Salt, RE/SPEC Technical Memorandum Report RSI-0068, RE/SPEC, Rapid City, SD, September 1978

(529) Parsons, Brinckerhoff, Quade and Douglas, Inc., Waste Isolation Facility Description-Bedded Salt, Y/OWI/SUB-76 16506, Office of Waste Isolation, Oak Ridge, TN, September 1976

(530) Stearns-Roger Engineering Company, Conceptual Design Report--National Waste Terminal Storage Repository for Storing Reprocessing Wastes in a Dome Salt Formation, DOE Contract No. EY-77-C-05-5367, U.S. Department of Energy, January 1979

(531) Kaiser Engineers, Inc., A National Waste Terminal Storagee Repository in a Bedded Salt Formation for Spent Unreprocessed Fuel, KE Report No. 78-57-RE, under DOE Contract No. EY-77-C-05-5366, U.S. Department of Energy, December 1978

(532) A.S. Burgess and P.O. Sandstrom, Immobilized Fuel Vault: Design Concepts and Layouts, Technical Memorandum Reports 5-1, pp. 18-19, Atomic Energy of Canada Limited, White Shell Nuclear Research Establishment, Pinawa, Manitoba, 1979

(533) T. Wiles and M.A. Mahtab, Immobilized Fuel Vault: Room-and-Pillar Thermal-Rock Mechanics Analyses, Technical Memorandum Reports 3-2, Atomic Energy of Canada Limited, White Shell Nuclear Research Establishment, Pinawa, Manitoba, 1979

(534) U.S. Environmental Protection Agency, A Workshop on Policy and Technical Issues Pertinent to the Development of Environmental Protection Criteria for Radioactive Wastes (three workshops): ORP-CSD-78-2, Denver, CO, March-April, 1978; ORP-CSD-77-1, Reston, VA, February 1977; ORP-CSD-77-2, Albuquerque, NM, 1977

(535) U.S. Nuclear Regulatory Commission, Proposed Goals for Radioactive Waste Management, NUREG-0300, 1978

(536) Office of Nuclear Waste Isolation, NWTS Program Criteria for Geologic Disposal of Nuclear Waste, Site Qualification Criteria, ONWI 33-2, Battelle Memorial Institute, Columbus, OH, 1980

(537) Pacific Northwest Laboratory, Reference Site Initial Assessment for a Salt Dome Repository, Vol. 1, PNL-2955, Richland, WA, 1979

(538) U.S. Department of Energy, Management of Commerically Generated Radioactive Waste - Draft Environmental Impact Statement, DOE/EIS-0046D (Vol. 1), 1979

(539) Panel on Geological Site Criteria, Committee on Radioactive Waste Management, Geological Criteria for Repositories for High-Level Radioactive Wastes, National Academy of Sciences, Washington, DC, August 1978

(540) 43 Fed. Reg. 53264, "U.S. Environmental Protection Agency Criteria," (November 15, 1978)

(541) R. LeStrange, A History of Herbal Plants, Arco Publishing Co., New York, NY, 1977

(542) G.E. Daniel, A Hundred and Fifty Years of Archeology, 2nd ed., Harvard University Press, Cambridge, MA, 1976

(543) H.C. Burkholder, Waste Isolation Performance Assessment -- A Status Report, ONWI-60, Battelle Memorial Institute, Columbus, OH, 1979

(544) S.E. Logan and M.C. Berbano, Development and Application of a Risk Assessment Method for Radioactive Waste Management, EPA 520/6-78-005, U.S. Environmental Protection Agency, 1978

(545) J.E. Campbell et al., Risk Methodology for Geologic Disposal of Radioactive Waste: Interim Report, SAND78-0029, Sandia National Laboratories, Albuquerque, NM, 1978

(546) R.A. Heckman and T.R. Donich, Waste Management Technical Support Project FY-78, End of Year Report, UCRL-52547, Lawrence Livermore Laboratory, Livermore, CA, 1978

(547) H.C. Burkholder et al., Waste Isolation Safety Assessment Program - Technical Progress Report for FY-77, PNL-2642, Pacific Northwest Laboratory, Richland, WA, 1979

(548) Arthur D. Little, Inc., Technical Support of Standards for High-Level Radioactive Waste Management, EPA-520/4-79-007, U.S. Environmental Protection Agency, 1979

(549) A. Brandstetter et al., Waste Isolation Safety Assessment Program -- Technical Progress Report for FY-78, PNL-2874, Battelle Northwest Laboratories, Richland, WA, 1979

(550) J.J. Jacobson, Chairman, Workshop on Potentially Disruptive Phenomena for Nuclear Waste Repositories, PNL-SA-7197, Pacific Northwest Laboratory, Richland, WA, 1979

(551) G.L. Benson, B.L. Scott, and M.A. Harwell, eds., A Summary of FY1978 Consultant Input for Task 1 - Release Scenario Analysis, PNL-2851, Battelle Northwest Laboratories, Richland, WA, 1979

(552) J.A. Stottlemyre, G.M. Petrie, and G.L. Benson, A Conceptual Simulation Model for Release Scenario Analysis of a Hypothetical Site in Columbia Plateau Basalts, PNL-2892 (Draft), Battelle Northwest Laboratories, Richland, WA, 1979

(553) J.A. Stottlemyre, G.M. Petrie, and M.W. Mullen, "Computer Enhanced Release Scenario Analysis for a Nuclear Waste Repository," Scientific Basis for Nuclear Waste Management, 1, p. 521, 1979

(554) H.C. Burkholder, "Waste Isolation Performance Assessment Program," Transactions of the American Nuclear Society, 33, p. 380, 1979

(555) R.W. Nelson and J.A. Schur, Methodology for Release Consequence Analysis -- Part I. PATHS Groundwater Flow Hydrologic Model, PNL-3162, Battelle Northwest Laboratories, Richland, WA, 1979

(556) A.E. Reisenauer, Assessment of Effectiveness of Geologic Isolation Systems, Variable Thickness Transient Ground-Water Flow Model, Vols. 1 and 2, PNL-3160-1/UC-70, PNL-3160-2/UC-70, Pacific Northwest Laboratory, Richland, WA, 1979

(557) R.T. Dillon, R.B. Lantz, and S.B. Pahwa, Risk Methodology for Geologic Disposal of Radioactive Waste: The Sandia Waste Isolation Flow and Transport (SWIFT) Model, SAND78-1267, Sandia National Laboratories, Albuquerque, NM, 1978

(558) S.K. Gupta, C.R. Cole, and F.W. Bond, Finite Element Three-Dimensional Groundwater (FE3DGW) Flow Model--Formulation, Program Listings, and Users' Manual, PNL-2939, Battelle Northwest Laboratories, Richland, WA, 1979

(559) C.R. Cole and S.K. Gupta, A Brief Description of the Three-Dimensional Finite Element Groundwater Flow Model Adapted for the Waste Isolation Safety Assessment Program, PNL-2652, Battelle Northwest Laboratories, Richland, WA, 1979

(560) Intera Environmental Consultants, Inc., Hydrology Model Evaluation at the Hanford Nuclear Waste Facility, ARH-C-00017, Atlantic Richfield Hanford Operations, Richland, WA, 1977

(561) Intera Environmental Consultants, Inc., Hydrogeological Analysis in Support of Waste Isolation Pilot Plant (WIPP) Program, report prepared for Sandia National Laboratories, Albequerque, NM, 1980

(562) T.A. Prickett, "Modeling Techniques for Groundwater Evaluation," Advances in Hydroscience, Vol. 10, 1975

(563) J.J. Fried, Groundwater Pollution, Else ier Publishing Company, New York, NY, 1975

(564) H.C. Burkholder and E.L.J. Rosinger, A Mo'el for the Transport of Radionuclides and Their Decay Products through Geologic Media, ONWI-11, Battelle Memorial Institute, Columbus, OH, 1979

(565) W.V. DeMier et al., GETOUT -- A Computer Program for Predicting Radionuclide Decay Chain Transport through Geologic Media, PNL-2970, Battelle Northwest Laboratories, Richland, WA, 1979

(566) J.F. Washburn, Methodology for Release Consequence Analysis -- Part IV Multicomponent Mass Transport Model, PNL-3179, Pacific Northwest Laboratory, Richland, WA, 1979

(567) D.B. McGlaughlin, Hanford Groundwater Modeling Statistical Methods for Evaluating Uncertainty and Assessing Sampling Effectiveness, RHO-C-18, Resource Management Associates, Lafayette, CA, January 1979

(568) Proceedings of the WISAP Task 4 Second Contractor Information Meeting, PNL-SA-7352, Battelle Northwest Laboratories, Richland, WA, October 1978

(569) J.B. Robertson, Numerical Modeling of Subsurface Radioactive Solute Transport from Waste -- Seepage Ponds at the Idaho National Engineering Laboratory, IDO-22057, Idaho Operations Office (AEC), Idaho Falls, ID, 1977

(570) Office of Nuclear Waste Isolation, Geologic Disposal of Radioactive Waste: Program Plan for Field Tests in Geologic Formations, ONWI-80 (1)-Draft, Battelle Memorial Institute, Columbus, OH, 1979

(571) T.D. Streltsova-Adams, "Well Hydraulics in Heterogeneous Aquifer Formations," Advances in Hydroscience, 11, 1978

(572) J.D. Duguid and R.C.Y. Lee, Water Resources Research, 13 (3), pp. 558-566, 1977

(573) J.F. Abel and J.O. Duguid, Finite Element Galerkin Method for Analysis of Flow in Fractural Porous Media, Finite Element Methods, pp. 599-615, U.A.H. Press, 1974

(574) A.S. Fari and P.A. Witherspoon. Numerical Simulation of Naturally Fractured Reservoirs, Paper S.P.E. 4290, presented at the S.P.E.-A.I.M.E. 3rd Symposium on Num. Sim. of Reservoir Performance, Houston, TX

(575) K.H. Coats, Geothermal Reservoir Modeling, Paper S.P.E. 6892, presented at the 52nd Annual Fall Meeting of the Society of Petroleum Engineers, A.I.M.E., Denver, CO, 1977

(576) J.E. Gale, A Numerical Field-Laboratory Study of Flow in Rocks with Informable Fractures - Scientific Series No. 72 - Inland Waters Directorate, Ottawa, Ontario, 1977

(577) H. Kazemi, "Pressure Transient Analysis of Naturally Fractured Reservoirs with Uniform Fractured Distribution," Soc. Pet. Eng. J., Vol. 246, pp. 451-462, 1969

(578) H. Kazemi et al., "Numerical Simulation of Water-Oil Flow in Naturally Fractured Reservoirs," Soc. Pet. Eng. J., pp. 317-326, December 1976

(579) E.J. Lefebre Du Prey and L.M. Weill, Frontal Displacement in a Fractured Reservoir, Finite Elem. Methods, pp. 653-671, U.A.H. Press, 1974

(580) T.D. Streltsova-Adams, "Fluid Flow in Naturally Fractured Reservoirs," 2nd Invitation Well Testing Synposium Proceedings, pp. 71-77, Berkeley, CA, October 25-27, 1978

(581) R.H. Rossen, "Simulation of Nat. Frac. Res. with Semi-implicit Power Terms," Soc. Pet. Eng. J., pp. 201-210, June 1977

(582) P.A. Witherspoon and O. Degerman, Swedish-American Cooperative Program on Radioactive Waste Storage in Mined Caverns in Crystalline Rock, Technical Project Report #1, Stripa LBL-7049 and SAC-01 and UC 70, Lawrence Berkeley Laboratory, Berkeley, CA, May 1978

(583) J.K. Osmond and J.B. Cowart, "The Theory and Uses of Natural Uranium Isotopic Variations in Hydrology," Atomic Energy Review, Vol. 14, No. 4, p. 621, IAEA, Vienna, 1976

(584) E.A. Bryant, OKLO, an Experiment in Long-Term Geologic Storage, Actinides in the Environment, ACS Symposium Series No. 35, American Chemical Society, New York, NY, 1976

(585) G.A. Cowen, Migration Paths for OKLO Reactor Products and Applications to the Problem of Geological Storage of Nuclear Wastes, LA-UR-77-2787, Los Alamos National Scientific Laboratory, Los Alamos, NM, 1977

(586) J.K. Sold t, ..M. Robinson, and D.A. Baker, Models and Computer Codes for Eval ing Environmental Radiation Doses, BNWL-1754, Battelle Northwest aboratories, Richland, WA, 1974

(587) D.A. Baker, G.R. Hoenes, and J.K. Soldat, FOOD: An Interactive Code to Calculate Internal Radiation Dose from Contaminated Food Products, BNWL-SA-5523, Battelle Northwest Laboratories, Richland, WA, 1976

(588) U.S. Nuclear Regulatory Commission, Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with Appendix I of 10 CFR, Part 50, Regulatory Guide 1.109, Revision 1, Appendix A, 1977

(589) J.R. Houston, D.L. Strenge, and E.C. Watson, DACRIN: A Computer Program for Calculating Organ Dose from Acute and Chronic Radionuclides Inhalation, BNWL-B389, Battelle Northwest Laboratories, Richland, WA, 1974

(590) D.L. Strenge and E.C. Watson, KRONIC: A Computer Program for Calculating Annual Average External Doses From Chronic Atmospheric Releases of Radionuclides, BNWL-B-351, Battelle Northwest Laboratories, Richland, WA, 1973

(591) D.L. Strenge, E.C. Watson, and J.R. Houston, SUBDOSA: A Computer Program for Calculating External Doses from Accidental Atmospheric Releases of Radionuclides, BNWL-B-351, Battelle Northwest Laboratories, Richland, WA, 1975

(592) U.S. Department of Energy, Management of Commercially Generated Waste, Appendix E, DOE/EIS-0046 (Draft), Washington, DC,

(593) R.G. Schrekhise, Model of Long-Term Accumulation of Radiocontaminants in Crop Plants, PNL-2636, Battelle Northwest Laboratories, Richland, WA, 1979

(594) Hansen, F.D., Quasi-Static and Creep Deformation Characteristics of Bedded Salt from the Carey Mine Near Lyons, Kansas, RE/SPEC Technical Memorandum Report RSI-0067 (Y/OWI/SUB-78/22303/13), Rapid City, SD, October 13, 1978

(595) H.S. Carslaw and J.C. Jaeger, Conduction of Heat in Solids, 2nd ed., Oxford University Press, New York, NY, 1959

(596) Tabulation of Waste Isolation Computer Models, ONWI/SUB-79/E512-04500 (Draft), Battelle Memorial Institute, Columbus, OH, November 1979

(597) H.C. Claiborne, W.C. Wagner, and R.A. Just, Comparison of Codes, ORNL/TM-7112, Oak Ridge National Laboratory, Oak Ridge, TN, December 1979

(598) W.D. Turner, D.C. Elrod, and I.I. Siman-Tov, HEATINGS - An IBM 360 Heat Conduction Program, ORNL/CSD/TM-15, Oak Ridge National Laboratory, Oak Ridge, TN, March 1977

(599) W.D. Turner, THAC-SIP-3D - A Three-Dimensional, Transient Heat Analysis Code Using the Strongly Implicit Procedure, K/CSD/TM-24, September 1978

(600) K.J. Bathe, ADINAT - A Finite Element Program for Automatic Dynamic Incremental Nonlinear Analysis of Temperatures, MIT-82448-5, Massachusetts Institute of Technology, Cambridge, MA, May 1977

(601) J.P. Smith, Systems Improved Numerical Differencing Analyzer User's Manual, NASA-CR-123271 (TRW 14690-H001-RD-00), April 1971

(602) A.L. Edwards, TRUMP: A Computer Program for Transient and Steady-State Temperature Distribution and Multidimensional Systems, UCRL-14754, Lawrence Livermore National Laboratory, Livermore, CA, Rev. 3, September 1972

(603) G.D. Callahan, Documentation of the Heat Transfer Code TRANCO, RE/SPEC Technical Memorandum Report RSI-0037 (ORNL/SUB-4269/15), Rapid City, SD, August 1975

(604) P.D. Fairchild and G. H. Jenks, Avery Island--Dome Salt In-Situ Test, Y/OWI/TM-55, Office of Waste Isolation, Oak Ridge, TN, June 1978

(605) J.L. Krumhansl, Final Report: Conasauga Near Surface Heater Experiment, SAND79-1855, Sandia Laboratories, Albuquerque, NM, November 1979

(606) M.T. Morgan, Thermal Conductivity of Rock Salt from Louisiana Salt Domes, ORNL/TM-6776, Oak Ridge National Laboratory, Oak Ridge, TN, June 1979

(607) Office of Waste Isolation, Thermal Analysis of a Ventilated High-Level Waste Repository, Y/OWI/SUB-76/16527, Oak Ridge, TN, April 1977

(608) H.C. Claiborne, L.D. Rickertsen, and R.F. Graham, Expected Environments for Spent Fuel/Waste Repository in Salt, ORNL/TM-7201 (Draft), Oak Ridge National Laboratory, Oak Ridge, TN, January 1980

(609) R.E. Westerman, The Preliminary Conceptual Design for Advanced Packages for the Geologic Disposal of Spent Fuel, PNL-2990, Pacific Northwest Laboratory, Richland, WA, 1979

(610) F.A. Kulacki et al., Experimental and Analytical Study of the Thermal Aspects of Deep Geological Disposal of Commercial Nuclear Wastes - Work Plan, ONWI/SUB79/E51203900/1, Battelle Memorial Institute, Columbus, OH, 1979

(611) Scientific Applications, Inc., Tabulation of Waste Isolation Computer Models, Vol. I, ONWI/SUB/79/E512-04500-1 (Draft), Oak Ridge, TN, November 1979

(612) G.D. Callahan, Documentation of the Heat Conduction Code TRANCO, RE/SPEC Technical Memorandum Report RSI-0037 (ORNL/SUB-4269/15), Rapid City, SD, August 1975

(613) G.D. Callahan, Documentation of the Thermal/Viscoelastic Code TEVCO, RE/SPEC Technical Memorandum Report RSI-0038 (ORNL/SUB-4269/17), Rapid City, SD, September 1975

(614) P.F. Gnirk and G.D. Callahan, Analytical Approximations to the Thermoelastic Behavior of Repository Configurations, RE/SPEC Technical Memorandum Report RSI-0064 (Y/OWI/SUB-78/22303/10), Rapid City, SD, January 1978

(615) R.A. Wagner, Quality Assurance Study of the Special Purpose Finite Element Program SPECTROM: I. Thermal, Elastic, Thermoelastic, and Viscoelastic Problems, RE/SPEC Topical Report RSI-0082 (ONWI/SUB-79/E512-02300/8), Rapid City, SD, April 19, 1979

(616) J.L. Ratigan and G.D. Callahan, Evaluation of the Predictive Capability of the Finite Element Method, RE/SPEC Technical Memorandum Report RSI-0065 (Y/OWI/SUB-78/22303/11), Rapid City, SD, August 17, 1978

(617) N.L. Carter, Petrofabric Analy_.s of the Deformational Behavior of Lyons, KS and Jefferson, LA Rock Salt, RE/SPEC Technical Memorandum Report RIS-0063 (Y/OWI/SUB-78/ 22303), Rapid City, SD, December 1977

(618) G.D. Callahan and J.L. Ratigan, Thermo/Viscoelastic Analysis of Waste-Container Sleeve: I, Time-Dependent Salt Loading on the Sleeve and Closure of the Drillhole, RE/SPEC Technical Memorandum Report RSI-0020 (ORNL/SUB4269/4), Rapid City, SD, October 1975

(619) G.D. Callahan and J.L. Ratigan, Thermo/Viscoelastic Analysis of Waste-Container Sleeve: II, Stress and Displacement Fields for the Room-and-Pillar Configuration, RE/SPEC Technical Memorandum Report RSI-0021 (ORNL/SUB-4269/5), Rapid City, SD, October 1975

(620) R. Hofmann, STEALTH, A Lagrange Explicit Finite-Difference Code for Solids, Structural, and Thermohydraulic Analysis, EPRI NP-260, Vol. 1, Users Manual, Science Applications, Inc., San Leandro, CA. Prepared for Electric Power Research Institute, Palo Alto, CA, Rev. 1, 1977

(621) D.E. Maxwell, Review and Continuing Analysis of the SAI Creep Model, ONWI/SUB-78/E412-01300, Scientific Applications, Inc., San Leandro, CA, December 1978

(622) D.E. Maxwell, Continuing Support of Rock Salt Model (Phase 2), ONWI/SUB-79/E512-01300/6, Scientific Applications, Inc., San Leandro, CA, October 1979

(623) R. Hofmann, B.W. Dial, and K.K. Wahi, Three-Dimensional Simulation of Project Salt Vault, OWNI/SUB/79/E512-01300/2, Scientific Applications Inc., San Leandro, CA, January 1979

(624) J.E. Gale and P.A. Witherspoon, An Approach to the Fracture Hydrology at Stripa: Preliminary Results, Technical Information Report No. 15, Swedish-American Cooperative Program on Radioactive Waste Storage in Mined Caverns in Crystalline Rock

(625) J.S.Y. Wang et al., A Study of Regional Temperature and Thermohydrological Effects of an Underground Repository for Nuclear Waste in Hard Rock, LBL-8271, Lawrence Berkeley Laboratory, Berkeley, CA, October 1978

(626) R. Hart and R. Hofmann, Hydrological Model in STEALTH 2D Code, ONWI/SUB/79/E512-01300/8, Battelle Memorial Institute, Columbus, OH, October 1979

(627) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository, DOE/TIC-11033 (Draft), April 1980

(628) N.G.W. Cook and M. Hood, Full-Scale and Time-Scale Heating Experiments at Stripa: Preliminary Results, Technical Project Report No. 11, Swedish-American Cooperative Program on Radioactive Waste Storage in Mined Caverns in Crystalline Rock, December 1978

(629) T.R. Anthony and H.E. Cline, Fourth International Symposium on Salt, Vol. I, p. 313, A.H. Cooper, ed., The Nothern Ohio Geological Society, Cleveland, OH, 1974

(630) P.F. Gnirk, W.B. Krause, and A.F. Fossum, State-of-the-Art Review of Brine Migration Studies in Salt, RE/SPEC Technical Memorandum Report RSI-0075, Rapid City, SD, February 1979 (revised January 1980)

(631) R.E. Gephart et al., Hydrologic Studies Within the Columbia Plateau, Washington--An Integration of Current Knowledge, RHO-BWI-ST-5, Rockwell Hanford Operations, Richland, WA, October 1979

(632) M.P. Hardy and G. Hocking, Numerical Modeling of Rock Stresses Within a Basaltic Nuclear Waste Repository, Phase II. Parametric Design Studies, RHO-BWI-C-23, University of Minnesota and Dames & More, July 1978

(633) M.S. Ayatollaki, Stress and Flow in Fractured Porous Media, Ph.D. Thesis, Department of Civil Engineering, University of California at Berkeley, Berkeley, CA, 1978

(634) J.W. Braithwaite and M.A. Molecke, Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation, SAND79-1935J, Sandia Laboratories, Albuquerque, NM, 1980

(635) M.O. Cloninger, A Perspectives Analysis on the Use of Engineered Barriers for Geologic Isolation of Spent Fuel, PNL-SA-7920, Battelle Northwest Laboratories, Richland, WA, 1979

(636) R.G. Dosch and G.W. Lynch, Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico, SAND78-0297, Sandia National Laboratories, Albuquerque, NM, 1978

(637) U.S. Department of Energy, Waste Isolation Pilot Plant-Draft Environmental Impact Statement, DOE/EIS-0026-D, 1979

(638) R.J. Serne, Batch K_d Measurements of Nuclides to Estimate Migration Potential at the Proposed Waste Isolation Pilot Plant in New Mexico, PNL-2448, Battelle Northwest Laboratories, Richland, WA, 1977

(639) J. Duguid and M. Reeves, Material Transport Through Porous Media, ORNL-4928, Oak Ridge National Laboratory, Oak Ridge, TN, 1975

(640) M. Reeves and J. Duguid, Water Movement Through Saturated - Unsaturated Media, ORNL-4927, Oak Ridge National Laboratory, Oak Ridge, TN, 1975

(641) M.J. Bell, ORIGEN - The ORNL Isotope Generation and Depletion Code, ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, TN, May 1973

(642) W.W. Engle, Jr., ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering, K-1693, RSIC Report CCC-82, March 1967

(643) W.A. Rhodes and F.R. Mynatt, The DOT III Two-Dimensional Discrete Ordinates Transport Code, ORNL/TM-4280, Oak Ridge National Laboratory, Oak Ridge, TN, September 1973

(644) E.A. Jenne, ed., "Chemical Modeling in Aqueous Systems," ACS Symposium Series 93, American Chemical Society, Washington, DC, 1979

(645) J.A. Apps et al., "Theoretical and Experimental Evaluation of Waste Transport in Selected Rocks, Waste Isolation Safety Assessment Program, Task 4, Contractor Information Meeting Proceedings, R.J. Serne, ed., pp. 189-307, PNL-SA-6957, Pacific Northwest Laboratory, Richland, WA, 1978

(646) R.J. Silva, et al., "Theoretical Experimental Evaluation of Waste Transport in Selected Rocks, FY1978 Annual Report," Waste Isolation Safety Assessment Program, Task 4, Second Contractor Information Meeting Proceedings, R.J. Serne, ed., Vol. 1, pp. 98-185, PNL-SA-7352, Pacific Northwest Laboratory, Richland, WA, 1979

(647) J.A. Apps, The Simulation of Reactions Between Basalt and Groundwater, p. 240, UCID-8066, Lawrence Berkeley Laboratory, Berkeley, CA, 1978

(648) L.V. Benson et al., Basalt Alteration and Basalt Water Interaction in the Pasco Basin of Washington State, Report 8532, Lawrence Berkeley Laboratory, Berkeley, CA, 1978

(649) T.J. Wolery, Correlation of Chemical Equilibria, Ph.D. Thesis, Part 2, Northwestern University, Evanston, IL, 1978

(650) T.J. Wolery, Calculation of Chemical Equilibrium Between Aqueous Solution and Minerals: The EQ3/6 Software Package, p. 41, UCRL-52658, Lawrence Livermore National Laboratory, Livermore, CA, 1979

(651) J.C. Westfall, J.I. Zachary, and F.M.M. Morel, MINEQL - A Computer Program for the Calculation of Chemical Equilibrium Composition of Aqueous Solutions, Note No. 18, EPA Grant R803738, Department of Civil Engineering, Massachusetts Institute of Technology, Cambridge, MA, 1976

(652) A. Brandstetter et al., Waste Isolation Safety Assessment Program Technical Progress Report for FY78, pp. 4-73 to 4-80, PNL-2874, Battelle Northwest Laboratories, Richland, WA

(653) R.C. Routson and R.J. Serne, One Dimensional Model of the Movement of Trace Radioactive Solute Through Soil Columns: The PERCOL Model, BNWL-1718, Battelle Northwest Laboratories, Richland, WA, 1972

(654) S.W. Ahlstrom et al, Multicomponent Mass Transport Model: Theory and Numerical Implementation (Discrete-Parcel-Random-Walk Version), BNWL-2127, Battelle Northwest Laboratories, Richland, WA, 1977

(655) R.J. Serne et al, Waste/Rock Interactions Technology Program (WRIT) Technology Program Plan for FY1980, Battelle Northwest Laboratories, Richland, WA, 1980

(656) N. Edelstein et al, Thermodynamic Properties of Chemical Species in Nuclear Waste - FY79 Work, Lawrence Berkeley Laboratory report to ONWI based on subcontract MPO No. 511-05100, Battelle Memorial Institute, Columbus, OH, 1979

(657) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository, DOE/TIC-11033 (Draft), April 1980

(658) M.O. Cloninger, A Perspective Analysis on Isolation of Spent Fuel, PNL-SA-7920, Battelle Northwest Laboratories, Richland, WA, September 1979

(659) H.C. Claiborne and F. Gera, Potential Containment Failure Mechanisms and Their Consequences at a Radioactive Waste Repository in Beded Salt in New Mexico, ORNL-TM-4639, Oak Ridge National Laboratory, Oak Ridge, TN, 1974

(660) H.C. Burkholder et al., "Incentives for Partitioning High-Level Waste," Nuclear Technology, 31, p. 202, 1976

(661) F. Girardi, G. Bertozzi, and M. D'Alessandro, Long Term Risk Assessment of Radioactive Waste Disposal in Geological Formations, EUR 5902.e, Commission of the European Communities, Ispra, Italy, 1977

(662) G. de Marsily et al., "Nuclear Waste Disposal: Can the Geologist Guarantee Isolation?" Science, 197, p. 519, 1977

(663) M. D. Hill and P. H. Grimwood, Preliminary Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste in Geologic Formations, NRPB-R69, National Radiological Protection Board, Harwell, U.K., 1978

(664) Nuclear Fuel Safety Group, Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste, Karn-Bransle-Sakerhet, Stockholm, Sweden, 1978

(665) L.E. Berman et al., Analysis of Some Nuclear Waste Management Options, URCL-13917, Lawrence Livermore National Laboratory, Livermore, CA, 1978

(666) U.S. Department of Energy, Environmental Aspects of Commercial Radioactive Waste Management, DOE/ ET-0029-Draft, 1979

(667) Working Group 7, International Nuclear Fuel Cycle Evaluation, Release Consequence Analysis for a Hypothetical Geologic Radioactive Waste Repository in Hard Rock, INFCE/DEP/WG7/21, December 1979

(668) M.A. Harwell et al., Reference Site Initial Assessment for a Salt Dome Repository, PNL-2955, (Working Document), Battelle Northwest Laboratories, Richland, WA, December 1979

(669) J.B. Robertson, Digital Modeling of Radioactive and Chemical Waste Transport in the Snake River Plateau Aquifer at the National Reactor Testing Station, Idaho, IDO-22054, U.S. Geological Survey, Idaho Falls, ID, 1974

(670) U.S. Department of Energy, Waste Isolation Pilot Plant - Draft Environmental Impact Statement, p. 9-120, DOE/E13-0026-D, 1979

(671) Working Group 7, International Nuclear Fuel Cycle Evaluation, Release Consequence Analysis for a Hypothetical Geologic Radioactive Waste Repository in Hard Rock, INFCE/DEP/WG7/21, pp. 4-5, December 1979

(672) Ibid., p. 2

(673) Ibid., p. 10

(674) Ibid., Figs. 7-8

(675) Ibid., Fig. 6

(676) Ibid., p. 5

(677) Ibid., p. 14

(678) Ibid., p. 18

(679) Ibid., Table XVIII

(680) Ibid., Fig. 12

(681) Ibid., Table XI

(682) Ibid., p. 21

(683) Ibid., Table XIX

(684) Ibid., Table XXV

(685) Bechtel National Inc., National Waste Terminal Storage (NWTS) Conceptual Reference Repository Description (CRRD), Vol. III, pp. 2.1-3 to 2.1-8, ONWI/SUB/79/E51201600.16 (Draft), Battelle Memorial Institute, Columbus, OH, September 1979

(686) Ibid., pp. 2.1-1 to 2.1-4

(687) Ibid., Vol. I, p. 3.4-6

(688) M.A. Harwell et. al., Reference Site Initial Assessment for a Salt Dome Repository, pp. 5-8 to 5-17, PNL-2955 (Working Document), Battelle Northwest Laboratories, Richland, WA, December 1979

(689) Ibid., p. 10-20

(690) Ibid., p. 10-23

(691) Ibid., p. 5-12

(692) Ibid., p. 5-15

(693) Ibid., pp. 5-8 to 5-17

(694) Ibid., p. 7-11

(695) Ibid., App. B

(696) Ibid., App. C

(697) Ibid., App. D

(698) Ibid., pp. 11-47 to 11-49

(699) W.C. Arcieri, J.G. Feinstein, and J.D. Freeman, Methods Used to Compute the Dose to Individuals and the General Population Due to Geotransport of Radionuclides from a Spent-Fuel Repository, NUS-TM-327 (Draft Report), NUS Corporation, Rockville, MD, February 1980

(700) M.A. Harwell et al., Reference Site Initial Assessment for a Salt Dome Repository, PNL-2955, (Working Document), Battelle Northwest Laboratories, Richland, WA, December 1979

(701) Ibid., Apps. G, H

(702) Ibid., pp. 11-47 to 11-49

(703) Y.S. Kim and J.G. Feinstein, Criticality Analysis for a Brine Filled Cavity in a Spent Fuel Nuclear Waste Repository Located in a Domed Salt Formation, NUS-TM-326 (Draft Report), NUS Corporation, Rockville, MD, January 1980

(704) Ibid., pp. 1-4

(705) Ibid., pp. 4-7

(706) M.A. Harwell et. al., Reference Site Initial Assessment for a Salt Dome Repository, pp. 4-8 to 4-9, PNL-2955 (Working Document), Pacific Northwest Laboratory, Richland, WA, December 1979

(707) Ibid., pp. 4-6 to 4-7

(708) Ibid., p. 4-7

(709) Ibid., p. 4-26

(710) Ibid., p. 4-17

(711) Ibid., p. 4-20

(712) Ibid., pp. 4-9 to 4-11

(713) Ibid., p. 4-27

(714) Ibid., pp. 4-20 to 4-26

(715) B.R. Erdal, R.D. Aguilar, B.P. Bayhurst, W.R. Daniels, C.J. Duffy, F.O. Lawrence, S. Maestas, P.Q. Oliver, and K. Wolfsberg; Sorption-Desorption Studies on Granite, LA-7456MS, Los Alamos National Scientific Laboratory, Los Alamos, NM, Feburary 1979

(716) W.R. Wawersik and D.W. Hannum, Interim Summary of Sandia Creep Experiments on Rock Salt from the WIPP Study Area, Southeastern New Mexico, SAND78-0115, Sandia National Laboratories, Albuquerque, NM, 1979(TSI)

(717) F.D. Hansen, Triaxial Quasi-Static Compression and Creep Behavior of Bedded Salt from Southeastern New Mexico, RE/SPEC Technical Memorandum Report RSI-0055, SAND79-7045, Sandia National Laboratories, Albuquerque, NM, November 1979(TSI)

(718) T.A. Duffy, The Salt Block I Test: Experimental Details and Comparison with Theory, SAND79-7050, Sandia National Laboratories, Albuquerque, NM, February 1980

(719) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository, p. 160, DOE/TIC-11033 (Draft), April 1980

(720) Ibid., p. 189

(721) Ibid., p. 189

(722) G.H. Jenks and C.D. Bopp, Storage and Release of Radiation Energy in Salt in Radioactive-Waste Repository, ORNC-TM-4449, Oak Ridge National Laboratory, Oak Ridge, TN, January 1974

(723) T.F. Lomenick, Accelerated Deformation of Rock Salt at Elevated Temperatures and Pressure and Its Implication for High Level Radioactive Waste Disposal, ORNL-TM-2102, Oak Ridge National Laboratory, Oak Ridge, TN, March 1968

(724) W.C. McClain, Rock Deformations Resulting From Project Salt Vault, ORNL-TM-4381, Oak Ridge National Laboratory, Oak Ridge, TN, November 1973

(725) E. Albrecht et al., "Disposal of Radioactive Wastes by Storage in a Salt Mine in the Federal Republic of Germany; Management of Low- and Intermediate-Level Radioactive Wastes, Proc. Symp. Aix-en-Provence 1970, IAEA, Vienna, Austria, STI/PUB/264, 753, 1970

(726) K. Batsche, W. Ravert, and K. Klair, Hydrological Research at the Site of the Asse Salt Mine, IAEA-SM-243/13, International Symposium on the Underground Disposal of Radioactive Wastes, July 1979

(727) K. Kuehn, "German In-Situ Studies in Salt," Proc. of the US/FRG Bilateral Workshop on Waste Isolation Performance Assessment and In-Situ Testing, ONWI-88, published by the Hahn-Meitner-Institut fuer Kernforschung, Berlin, FRG, October, 1979

(728) Technical Exchange and Cooperative Arrangement between U.S. AEC and the Federal Ministry for Research and Technology of the FRG in the Field of Management of Radioactive Wastes, December 1974

(729) N.E. Carter (ONWI), letter to J.O. Neff (DOE/RL-C), discussing "Joint US/FRG Cooperative Program Recommendations," January 1980

(730) C.L. Christensen, Test Plan, Bell Canyon Test WIPP Experimental Program, Borehole Plugging, SAND79-1305, Sandia National Laboratories, Albuquerque, NM, June 1979

(731) T.O. Hunter, Summary of Research and Development Activities in Support of Waste Acceptance Criteria for WIPP, SAND79-1305, Sandia National Laboratories, Albuquerque, NM, December 1979

(732) A.R. Sattler, T.O. Hunter, and H.C. Walker, In-Situ Experiments to Support Development of WIPP, Presented at the Materials Research Society Conf., Boston, MA, November, 1979, SAND79-0905C, Sandia National Laboratories, ALbuquerque, NM.

(733) T.O. Hunter, Summary of Results and Plans of the Experimental Program for the Waste Isolation Pilot Plant, SAND79-1950C, Sandia National Laboratories, Albuquerque, NM, October 1979

(734) C.W. Cook and E.S. Ames, Borehole Inclusion Stressmeter in Bedded Salt, SAND79-0363, Sandia National Laboratories, Albuquerque, NM, 1979

(735) W.E. Wowak and A.R. Sattler, Criteria and Preliminary Design for WIPP Exploratory Program Experiments, SAND79-20077, Sandia National Laboratories, Albuqueque, NM, 1980

(736) A.R. Sattler and T.O. Hunter, Pre-WIPP Experiments in Salt: Part I: Executive Summary; Part II: Program Description, SAND79-0025, Sandia National Laboratories, Albuquerque, NM, 1979

(737) W.B. Krause, Experimental Plan for In-Situ Synthetic and Natural Brine Movement Studies at Avery Island, RE/SPEC Topical Report RSI-0083, (ONWI/SUB-79/E512-02200), RE/SPEC Rapid City, SD, July 1979

(738) W.B. Krause and P.F. Gnirk, "Dome-Salt Brine-Migration Experiments at Avery Island, Louisiana, Proceedings of the NWTS Program Information Meeting, ONWI-62, Battelle Memorial Institute, Columbus, OH, October 1979

(739) R.I. Ewing, WIPP Test of a Radiant Heater in the Avery Island Salt Mine, Sandia National Laboratories, Albuquerque, NM, September 1979

(740) R.I. Ewing, Memo, "Test of WIPP Radiant Heater in the Avery Island Salt Mine: First Month of Operation," Sandia National Laboratories, Albuquerque, NM, December 1979

(741) T.O. Hunter, Test Plan for Scanning of Intact Salt with Radar at the Avery Island Mine, Sandia National Laboratories, Albuquerque, NM, December 1979

(742) A.R. Sattler and C.W. Cook, Scanning of Intact Salt at Avery Island Mine, News Release, Sandia National Laboratories, Albuquerque, NM, December 1979

(743) P.A. Witherspoon and O. Degerman, Swedish-American Program on Radioactive Waste Storage in Mine Caverns Program Summary, LBL-7019 and SAC-01, Lawrence Berkeley National Laboratory, Berkeley, CA, May 1978

(744) N. Cook and M. Hood, Full-Scale and Time Scale Heating Experiments at Stripa: Preliminary Results, LBL-7072 and SAC-11, Lawrence Berkeley National Laboratory, Berkeley, CA, December 1978

(745) L. Lundstrom and H. Stille, Large Scale Permeability Test of the Granite in the Stripa Mine and Thermal Conductivity Test, LBL 7052 and SAC-02, Lawrence Berkeley National Laboratory, Berkeley, CA, July 1978.

(746) H. Carlsson, Stress Measurement in the Stripa Granite, LBL-7078 and SAC-04, Lawrence Berkeley National Laboratory, Berkeley, CA, August 1978.

(747) P.A. Witherspoon, N.G.W. Cook and J.E. Gale, Geologic Storage of Radioactive Waste--Results from Field Investigations at Stripa, Sweden

(748) O. Landstrom, C.E. Klockars. and K.E. Holmberg, "In-Situ Experiments on Nuclide Migration in Fractured Crystalline Rocks," KBS Teknisk Rapport 110, Studsink Energiteknik and the Geologic Survey of Sweden, July 1978

(749) L.D. Ramspott, L.B. Ballou, R.C. Carlson, D.N. Montan, T.R. Butkovich, J.E. Duncan, W.C. Patrick, D.C. Wilder, W.G. Brough and M.C. Mayr, Technical Concept for a Test of Geologic Storage of Spent Reactor Fuel in the Climax Granite, Nevada Test Site, UCRL 52796, Lawrence Livermore National Laboratory, Livermore, CA, June 1979

(750) L.D. Ramspott and L.B. Ballou, Test Storage of Spent Reactor Fuel in the Climax Granite at the Nevada Test Site, Paper presented at Tucson Waste Management Conference Tucson, AR, March 1980

(751) G.H. Jenks, Effect of Gamma Radiation on Mechanical and Physical Properties of Crystalline Rock, ORNL TM-4927, Oak Ridge National Laboratory, Oak Ridge, TN, September 1975

(752) M.D. Voegele, "Heated Block Test in Granitic Gneiss," Technical Progress Report for the Quarter, July 1 to September 30, 1979, ONWI-9(4), Battelle Memorial Institute, Columbus, OH, 1979

(753) W. Hustohlid, "Crystalline Rock Mining Technology" Technical Progress Report for the Quarter October 1 to December 31, 1979, ONWI-9(5), Battelle Memorial Institute, Columbus OH, 1979

(754) D.F. McVey, A.R. Lappin and R.K. Thomas, The Results and Supporting Analysis of a Near Surface Heater Experiment in the Eleana Argillite, Presented at NEA/OECD Workshop on the Use of Argillaceous Materials for Isolation of Radioactive Wastes, Paris, France, September 1979

(755) A.R. Lappin and W.A. Olsson, Material Properties of Eleana Argillite Extrapolation to other Argillaceous Rock, and Implication for Waste Management, Presented at NEA/OECD Workshop on the Use of Argillaceous Materials for Isolation of Radioactive Waste, Paris, France, September 1979

(756) D.F. McVey, R.K. Thomas, and A.R. Lappin, Small Scale Heater Tests in Argillite of the Eleana Formation at the Nevada Test Site, SAND79-0344, Sandia National Laboratories, Albuquerque, NM, November 1979

(757) J.R. Krumhansl, Final Report: Conasauga Near-Surface Heater Experiment, SAND79-1855, Sandia National Laboratories, Albuquerque, NM, November 1979

(758) Rockwell Hanford Operations, Near Surface Test Facility Test Program (Phase I and Phase II), BWI 02TP0101, Richland, WA, August 1979

(759) A.D. Krug, "Basalt Near Surface Test Facility Test Plans," Proc. of a Workshop on Thermomechanical Modeling for a Hardrock Waste Repository, pp. 113-120, UCAR-10043, Lawrence Livermore National Laboratory/ONWI, Berkeley, CA, June 1979

(760) E.A. Bryant et al., "OKLO, an Experiment in Long-Term Geologic Storage," Actinides in the Environment, ACS Symposium Series No. 35, Paper 6, pp. 89-102, American Chemical Society, New York, NY, 1976

(761) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository, p. 186, DOE/TIC-11033 (Draft), April 1980

(762) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository, p. 187, DOE/TIC-11033 (Draft), April 1980

(763) 10 CFR, Part 20, "Standards for Protection Against Radiation"

(764) U.S. Nuclear Regulatory Commission, NRC Standard Review Plan 12.3, "Radiation Protection Design Features"

(765) 10 CFR, Part 50, "Licensing of Production and Utilization Facilities"

(766) 10 CFR, Part 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions"

(767) 10 CFR, Part 72 (Proposed), "Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)"

(768) 10 CFR, Part 70, "Domestic Licensing of Special Nuclear Material"

(769) U.S. Nuclear Regulatory Commission, Regulatory Guides Division 3, Fuels and Materials Facilities

(770) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as is Reasonable (Nuclear Power Reactors)"

(771) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable"

(772) Kaiser Engineers, Conceptual Design Report, National Waste Terminal Storage Repository for Storing Spent Fuel in a Bedded Salt Formation, "Occupational and Health Physics Studies," KE Report No. 78-21-R, Oakland, CA, September 1978

(773) Stearns-Roger Engineering Company, Conceptual Design Report, National Waste Terminal Storage Repository for Storing Reprocessing Wastes in a Dome Salt Formation, Volume 1, Denver, CO, 1979

(774) Stearns-Roger Engineering Company, National Waste Terminal Storage Repository No. 1 - Conceptual Design Description, Denver, CO, January 17, 1979

(775) Kaiser Engineers, Conceptual Design Report, National Waste Terminal Storage Repository in a Bedded Salt Formations, Vol. 1, KE Report 78-57-RE, Oakland, CA, 1978

(776) U.S. Department of Energy, Safety, Health, and Environmental Protection, Interim Management Directive 5001, September 29, 1977

(777) U.S. Nuclear Regulatory Commission, Safety Evaluation Report: General Electric Co., Midwest Fuel Recovery Plant, Docket No. 50-268, Washington, DC

(778) U.S. Nuclear Regulatory Commission, Safety Evaluation Report: Allied General Nuclear Services, Reprocessing Plant, Docket No. 50-332, Washington, DC

(779) U.S. Nuclear Regulatory Commission, Safety Evaluation Report: Nuclear Fuel Services, West Valley Reprocessing Plant, Docket No. 50-201, Washington, DC

(780) U.S. Nuclear Regulatory Commission, INREM-II: A Computer Implementation of Recent Models for Estimating the Dose Equivalent to Organs of Man From an Inhaled or Ingested Radionuclides, NUREG/CR-00114, June 1978

(781) International Commission on Radiological Protection, Task Group on Lung Dynamics, "Deposition and Retention Models for Internal Dosimetry of the Human Respiratory Tract," Health Physics, 12, p. 173-207, 1966

(782) U.S. Department of Energy, Draft Environmental Impact Statement - Waste Isolation Pilot Plant, "Chapter 9, Accident Scenario R-15," DOE/EIS-0026-D, April 1979

(783) "Recommendations of the International Commission on Radiological Protection," ICRP Publication 9, (Adopted September 17, 1965)

(784) U.S. Environmental Protection Agency, Manual of Protection Action Guides and Protective Actions for Nuclear Incidents, September 1975

(785) U.S. Nuclear Regulatory Commission, 10 CFR, Part 50, Appendix E, "Emergency Plans," Proposed Change, 44 Fed. Reg. 75167, 19 December 1979

(786) R.E. Stinebaugh, Retrieval of Canistered Experimental Waste at Waste Isolation Pilot Plant, SAND79-1239, Sandia National Laboratories, Albuquerque, NM, 1979

(787) U.S. Department of Energy, Safety Assessment Document for the Spent Reactor Fuel Geologic Storage Test in the Climax Granite Stock at the Nevada Test Site, NVO-210, Nevada Operations Office, Las Vegas, NM, January 1980

(788) U.S. Nuclear Regulatory Commission, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," Table 1: Acceptable Surface Contamination Levels

(789) R.L. Bradshaw and W.C. McClain, eds., Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, Chapter 2, ORNL-4555, Oak Ridge National Laboratory, Oak Ridge, TN, 1971

(790) U.S. Atomic Energy Commission, Environmental Statement, Radioactive Waste Repository, Lyons, KS, WASH-0116, 1971

(791) U.S. Energy Research and Development Administration, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, ERDA-76-43, Washington, DC, 1976

(792) S.E. Logan and M.C. Berbano, Development and Application of a Risk Assessment Method for Radioactive Waste Management, EPA 510/6-78-005, U.S. Environmental Protection Agency, 1978

(793) Electric Power Research Institute, Status of Commercial Nuclear High-Level Waste Disposal, EPRI NP-44-SR, Palo Alto, CA, 1976

(794) J. E. Campbell, et. al., Risk Methodology for Geologic Disposal of Radioactive Waste: Interim Report, SAND78-0029, Sandia National Laboratories, Albuquerque, NM, 1978

(795) Committee on Radioactive Waste Management, National Research Council, An Evaluation of the Concept of Storing Radioactive Wastes in Bedrock Below the Savannah River Plant Site, National Academy of Sciences, Washington, DC, 1972

(796) International Atomic Energy Agency, Management of Radioactive Wastes from the Nuclear Fuel Cycle: Proceedings of a Symposium, Organized by the International Atomic Energy Agency and the OECD Nuclear Energy Agency, Vienna, Austria, 1976

(797) M. Bergman, International Symposium on Storage in Excavated Rock Caverns: Proceedings of First International Symposium, held in Stockholm, Sweden, September 1977, Pergamon Press, Oxford-New York, 1978

(798) National Academy of Sciences - National Research Council Committee on Radioactive Wastes Management, Geological Criteria for High-Level Radioactive Wastes/ Panel on Geological Site Criteria, National Research Council, Washingon, DC, 1977

(799) G. D. Brunton and W.C. McClain, Geological Criteria for Radioactive Waste Repositories, Y/OWI/TM-47, Office of Waste Isolation, Oak Ridge, TN, 1977

(800) Nuclear Regulatory Commission, Workshop for State Review of Site Suitability Criteria for High-Level Radioactive Waste Repository, NUREG-0354, February 1978

(801) Environmental Protection Agency, "Criteria for Radioactive Wastes," Fed. Reg., Part IX, Washington, DC, November 15, 1978

(802) 40 CFR, Subchapter F, "Radiation Programs," Part 190

(803) 40 CFR, Subchapter C, "Air Programs," Parts 60, 61

(804) 40 CFR, Subchapter D, "Water Programs"

(805) U.S. Department of Energy, Waste Isolation Pilot Plant (WIPP)-Draft Environmental Impact Statement, Vol. 1, DOE/EIS-0026-D, 1979

(806) U.S. Department of Energy, Management of Commercially Generated Radioactive Waste - Draft Environmental Impact Statement, Vol. 1, pp. 3.1.98 to 3.1.226, DOE/EIS-0046-D, 1979

(807) Ibid., p. 1.2

(808) Ibid., p. 1.7

(809) Ibid., p. 1.11

(810) Ibid., p. 1.19

(811) Ibid., p. 3.1.100

(812) Ibid., p. 3.1.105

(813) Ibid., pp. 3.1.216, 3.1.218, 3.1.220, 3.1.222

(814) Ibid., p. 3.1.127

(815) U.S. Department of Energy, Management of Commercially Generated Radioactive Waste - Draft Environmental Impact Statement, Vol. I, p. 3.1.136ff, DOE/EIS-0046-D, Washington, DC, 1979

(816) Ibid., p. 3.1.155

(817) Ibid., p. 3.1.136

(818) U.E. Shelford, The Ecology of North America, University of Illionis Press, Urbana, IL, 1963

(819) Ibid., p. 3.1.116, Table 3.1.11

III PROGRAM FOR ESTABLISHING MINED GEOLOGIC REPOSITORIES

III.A INTRODUCTION

Part II demonstrates the technical basis for concluding that spent nuclear fuel can be disposed of in mined geologic repositories in a safe and environmentally acceptable manner. Part III describes the Department's program for ensuring that such repositories will be available in a reasonable time frame and at a reasonable cost.

Chapter III.B outlines the organization that the Department has established to implement the National Waste Terminal Storage (NWTS) Program.

Chapter III.C describes major decisions required to implement the mined geologic disposal option. These are associated with the processes of site selection with State consultation and concurrence and of repository licensing, both of which affect the timing of the NWTS Program. The Department herein demonstrates that candidate sites will be selected by a systematic process that includes consideration of all applicable factors. This process will be conducted with participation of the public and involvement of State and local governments, as specified in the President's message of 12 February 1980 (1). The impacts of the anticipated NRC licensing process on the repository development schedule also are described.

Chapter III.D discusses a number of factors germane to the schedule of mined geologic repository development and explains how the NWTS Program addresses each of these factors. These factors include the following:

1. Implemention of the National Environmental Policy Act of 1969.
2. Cooperation of multiple Federal agencies.
3. Land acquisition activities.
4. Availability of expert staff.
5. Logistics and administration.
6. Design and construction time.

7. Waste retrieval period and backfill.

8. Technology development.

Chapter III.E presents estimates of the cost of mined geologic disposal of spent fuel. Costs for research and development and for repository siting, construction, and operation are identified. Finally, Chapter III.F provides an integrated overview of the Department's proposed schedules and costs.

III.B PROGRAM ORGANIZATION AND MANAGEMENT

III.B.1 Department of Energy Organization

The President has given the Secretary of Energy overall responsibility for integrating the Nation's nuclear waste management program (1). The Secretary of Energy has the lead role for (i) coordinating all Federal nonregulatory aspects of nuclear waste management; (ii) maintaining effective working relationships with regulatory bodies, such as the Environmental Protection Agency (EPA) and the Nuclear Regulatory Commission (NRC); and (iii) developing strong and effective ties between the Federal Government and the States on all aspects of nuclear waste management.

Within the Department of Energy, nuclear waste management program activities are under the direction of the Assistant Secretary for Nuclear Energy who reports to the Undersecretary and the Secretary. The Deputy Assistant Secretary for Nuclear Waste Management directs the Office of Nuclear Waste Management (ONWM) and is responsible for managing all aspects of the Department's management programs both for disposal (the NWTS Program) and for storage (described in V.C.1). The relationship of the NWTS Program management structure with the Department and with other Federal agencies is shown in Figure III-1.

Within ONWM, the Director of the Division of Waste Isolation is responsible for overall direction of the NWTS Program. This division implements the objectives of the NWTS Program by directing and controlling activities, including budgetary allocations, of various Department field offices and contractors.

To effectively implement the NWTS Program and other programs of this magnitude, the Department has decentralized the program activities. Under decentralization, the Department headquarters personnel are responsible for the development of overall plans, establishment of priorities, and analysis of program requirements. The management of resources to accomplish a given objective within prescribed monetary limits and schedules is the responsibility of the field operations offices.

Department field office personnel supported by over 2,000 professional employees of contractors are responsible for implementation of the NWTS program. This arrangement provides the program with experience of a broad spectrum of professionals ranging from geoscientists and mechanical engineers to sociologists and political scientists. Figure III-1 shows that technology development and characterization of sites are performed by contractors who report to field operations offices.

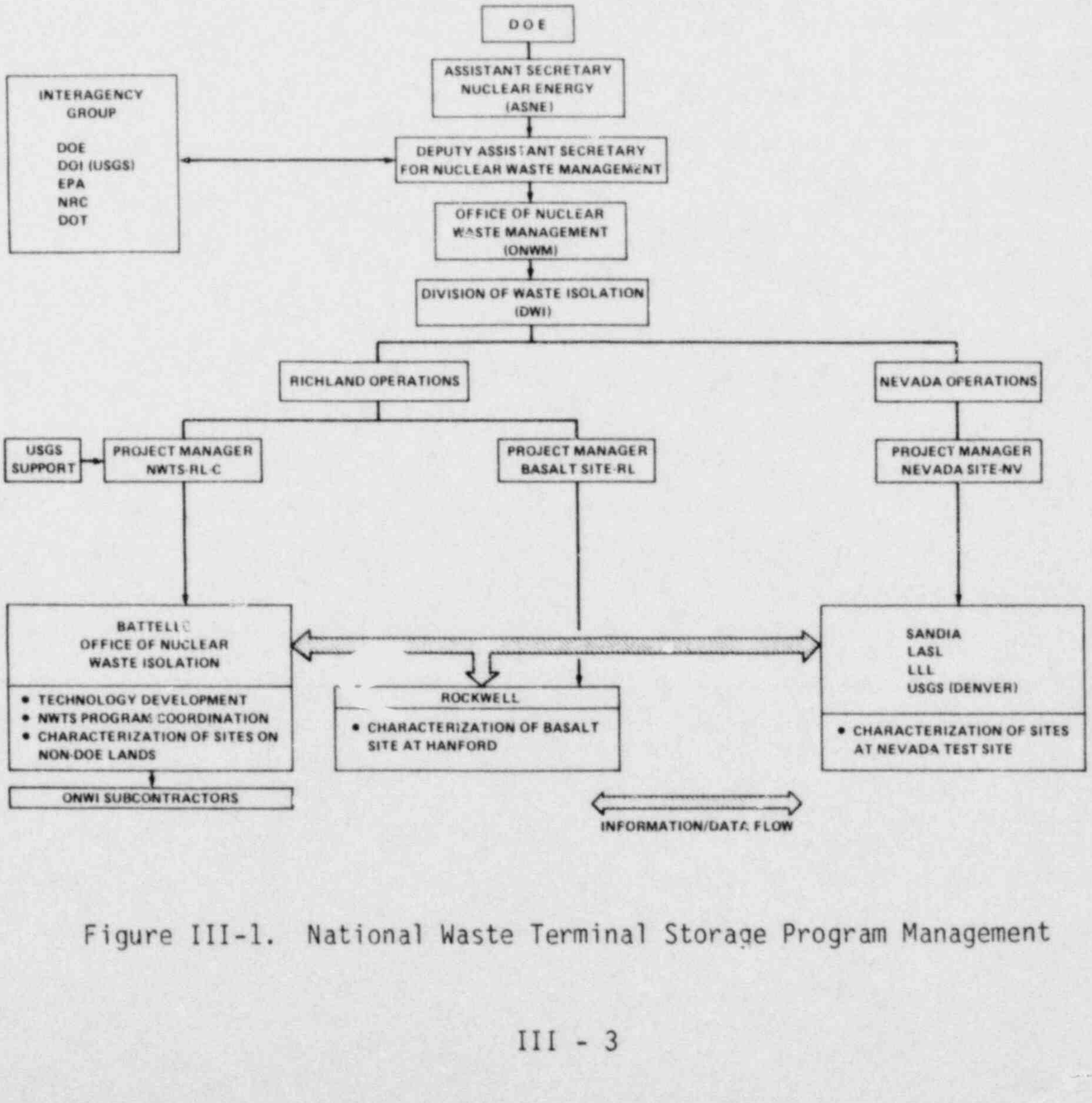


Figure III-1. National Waste Terminal Storage Program Management

Three NWTS projects are involved in the mined geological repository program: the Office of Nuclear Waste Isolation (ONWI), the Basalt Waste Isolation Project (BWIP), and the Nevada Nuclear Waste Storage Investigations (NNWSI). The interrelationship of these projects is shown in Figure III-1, with their respective supporting organizations: Battelle Memorial Institute; Rockwell International; and, at the Nevada Test Site (NTS), Sandia National Laboratory, Los Alamos National Laboratory, Lawrence Livermore Laboratory, and the U.S. Geological Survey. All three projects conduct independent site characterization and evaluation, leading to recommendations of preferred sites within their respective programs; they also share data and information of mutual benefit. In addition to project responsibilities, the Richland Operations Office, through its Columbus Program Office, oversees the ONWI responsibility for generic technology direction and coordination of site investigations in the NWTS Program (depicted by the shaded arrows in Figure III-1). Further details about the three main NWTS projects are contained in the following sections.

Also shown in Figure III-1 is an Interagency Working Committee on Radioactive Waste Management chaired by the Deputy Assistant Secretary for Nuclear Waste Management. This committee has been established to ensure that the President's Waste Management policy is properly implemented (see also III.D.2).

III.B.1.1 Office of Nuclear Waste Isolation

ONWI in Columbus, Ohio, has lead responsibility in the NWTS Program structure for coordinating all projects, for developing general waste disposal technology, and for geologic exploration and environmental studies conducted on non-Department lands. ONWI's responsibilities include making recommendations to the Department concerning site qualification criteria, safety assessment methodologies, licensing strategies, and the collection and evaluation of data for site and repository suitability assessment. For example, the criteria which will be used for site selection, now being circulated for external review and comment, were developed by ONWI and agreed upon by all three NWTS projects and the Division of Waste Isolation (2). Program planning and coordination of examination of alternate disposal concepts also are assigned to ONWI. ONWI's efforts are supervised by the Columbus Program Office of the Department's Richland Operations Office.

At the present time, the feasibility of locating repositories in various geologic formations throughout the continental United States is being evaluated under ONWI's direction. This evaluation conforms with the President's statement of 12 February 1980: "Immediate attention will focus . . . on locating and characterizing a number of potential repository sites, in a variety of different geologic environments with diverse rock types" (1).

III.B.1.2 Basalt Waste Isolation Project

The objective of the BWIP is to investigate the suitability of basalt formations for waste disposal and to evaluate the use of the Department's Hanford Site as a potential site for waste disposal. The Hanford Site, located in the Columbia River Plateau of southeastern Washington, is underlain by thick basalt formations. This reservation was selected for study because of its present commitment to nuclear activities, the possible continued dedication of Hanford to this purpose, and the potential of basalt as an acceptable isolation medium. Technology development studies, including the evaluation of engineered and natural barriers to waste migration, also are under way. A Near-Surface Test Facility is being constructed in a basalt outcropping at Hanford to carry out in-situ tests, including tests with placement of spent fuel. These tests will determine how nuclear wastes will interact with the host rock by studying the thermal, mechanical, and radiation effects on basalt, and provide engineering data to support the design and construction of a repository.

The Department's Richland Operations Office directs the management, quality assurance, contract administration, and data collection and evaluation of the project, with technical assistance provided by the Rockwell International Corporation.

III.B.1.3 Nevada Nuclear Waste Storage Investigations

The NNWSI project is being conducted at the Department's Nevada Test Site (NTS) in southern Nevada. The NTS provides a variety of geologic media such as granites, argillites (a compact clay rock), and tuff (a heat-fused volcanic ash). This site was selected for study because of its present commitment to nuclear activities and its geologic characteristics.

NNWSI's tasks include evaluating the area's possible use for a waste repository, identifying a preferred site in a suitable geologic medium for a repository, and in situ testing of thermal and radiation effects of encapsulated spent nuclear fuel placed underground in a granite formation. Other studies are evaluating the effectiveness of shale, granite, and tuff as host media and the impact, if any, of ground motion from weapons testing on repository design. This effort is managed by the Department's Nevada Operations Office with technical assistance provided by Sandia National Laboratory, Los Alamos National Laboratory, Lawrence Livermore Laboratory, and the U.S. Geological Survey.

III.B.2 Management

Program activities are planned, controlled, and executed by application of detailed management procedures. As stated earlier, NWTS projects are controlled by the Department's lead field offices and lead contractors under the overview and guidance of the Office of Nuclear Waste Management. A Program Plan is prepared annually by each NWTS component (3, 4, 5) and, when approved by the Department, is a basis for each fiscal year's program. When approved by the Department's lead field office and by the Division of Waste Isolation, the Program Plans represent an agreement as to how the project will be conducted and controlled. These Program Plans describe objectives, work category structures, schedules, budgets, and organizational accountabilities within the NWTS Program.

The implementation and effectiveness of project control are monitored by the responsible Department field office and the Division of Waste

Isolation through periodic project reviews and progress reports. The project lead contractor is required to conduct mid-year and annual program reviews at which the Department's managers assess the adequacy and effectiveness of individual NWTS activities. In addition, contractors are required to submit monthly and quarterly progress reports which provide NWTS management with the means and opportunity to assess accomplishments. Redirection can be initiated as a result of any of these reviews or reports. Field office directives provide continuous direction or redirection, if required, to the lead contractor, supplementing his initial contractual requirements.

Although control of the day-to-day activities of each of the NWTS projects is achieved independently from the other projects, integration of the projects is accomplished by periodic meetings of the field office project managers and the Director of the Division of Waste Isolation. The Department field office manager exercises direct project control through routine meetings with contractor personnel.

Additional management and administrative processes, e.g., peer review and technical consultation, planning and control techniques, quality assurance procedures, configuration and data management approaches, and reporting and information control methods, are routinely used in the NWTS Program.

III.C MAJOR DECISIONS REQUIRED TO IMPLEMENT THE MINED GEOLOGIC DISPOSAL OPTION

The schedule for implementation of the geologic disposal option depends on the following major decisions:

1. Site selection with State consultation and concurrence.
2. Licensing for construction of the repository.

These are the focus of preparatory technical activities that must be conducted in cooperation with those agencies, organizations, and individuals outside of the Department which participate in decisionmaking.

This chapter describes the processes leading to a successful completion of the NWTS Program. The relevance of each decision to implementation of geologic disposal is discussed, as are the processes that lead to each decision. An integrated presentation of the milestones discussed in Chapters III.C and III.D is provided in Figures III-2 and III-3. Figure III-2 presents a reference schedule composed of milestones leading to the earliest dates for repository operation. Figure III-3 presents an extended schedule, incorporating longer durations for the site characterization and selection phase, and the licensing and construction phases of the program to allow for delays that may occur. These schedules and forecasts of repository availability dates are discussed in further detail in Chapter III.F.

III.C.1 Selection of Candidate Sites For Repositories

III.C.1.1 Introduction

As directed in the President's statement of 12 February 1980 (1), the Department intends to identify candidate sites at several locations and in different media before recommending a specific site for the first license application. Candidate sites will be selected by a systematic process, taking into consideration all applicable factors. Details of the Department's program leading to selection of a site will be proposed in a forthcoming NWTS site characterization and selection plan (6), which includes three phases:

1. Site exploration, characterization, and banking.
2. Detailed site characterization.
3. Site selection.

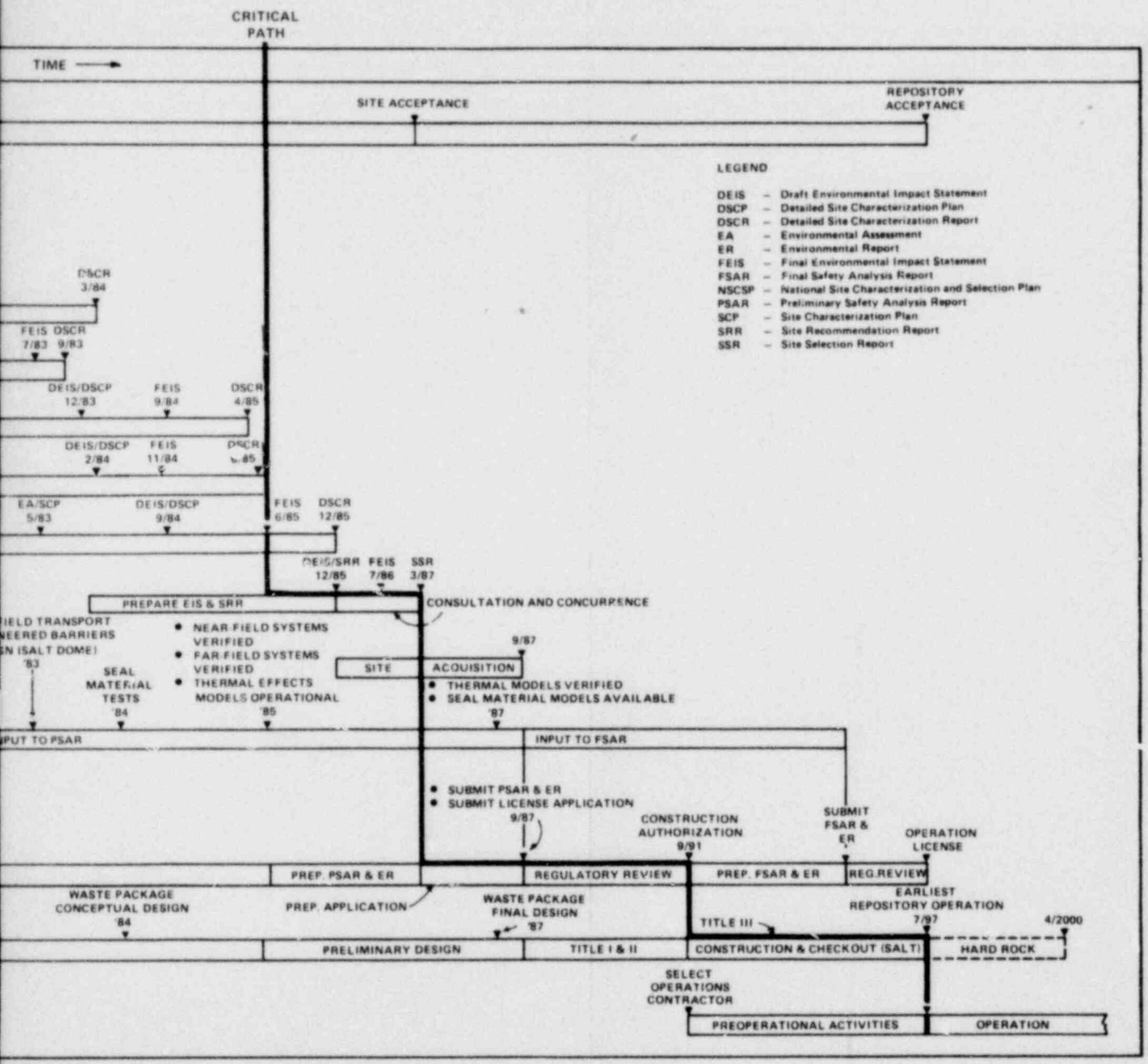
The various activities included in these phases are shown in Figure III-4.

Exploration and characterization of surface and subsurface environments progress through increasingly detailed studies of narrowed geographical and geologic areas until preferred sites are identified. Detailed site characterization is then undertaken to complete the data needed for preparation of licensing documents.

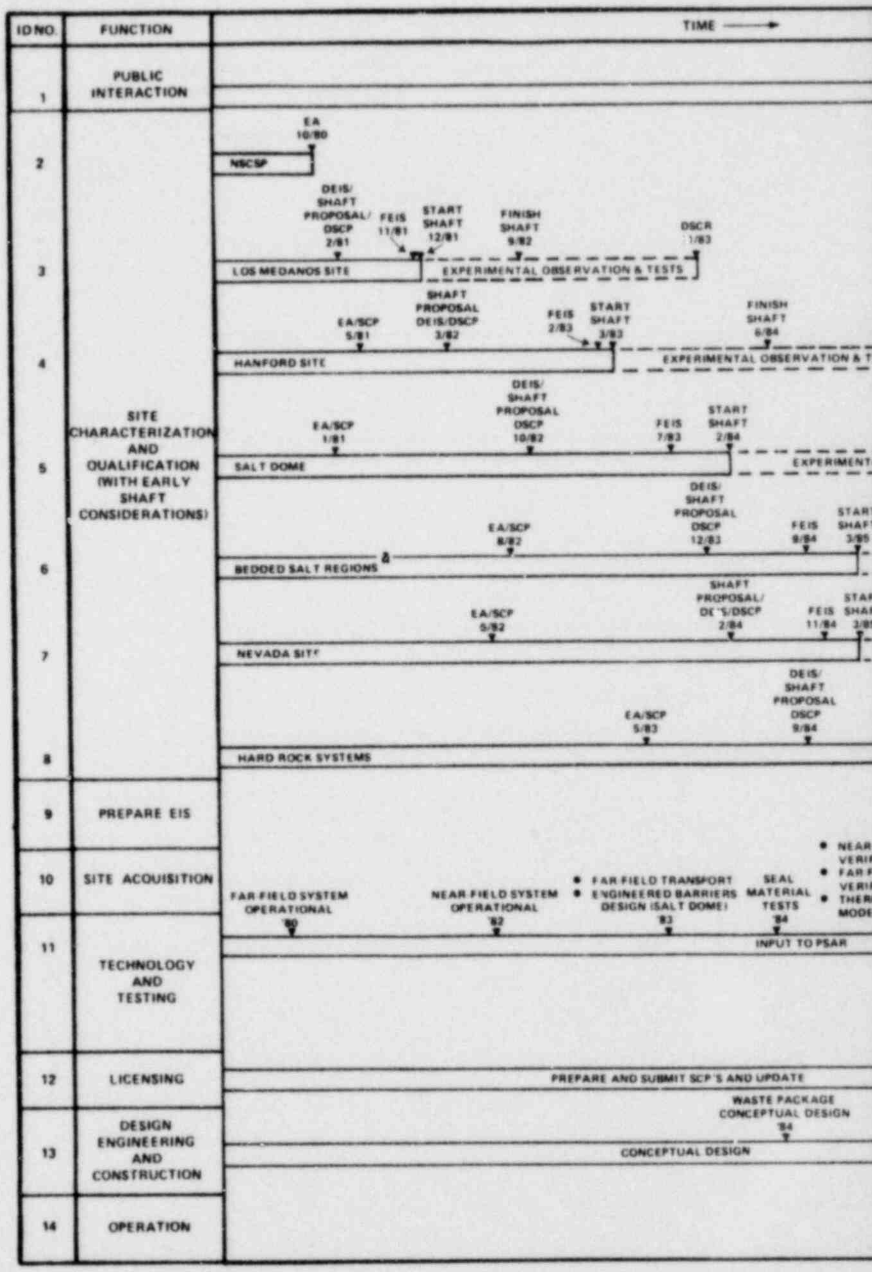
ID NO.	FUNCTION	
1	PUBLIC INTERACTION	
2	SITE CHARACTERIZATION AND QUALIFICATION	NSCSP; EA 10/80
3		LOS MEDANOS SITE; DEIS 2/81; FEIS 11/81; DSCR 8/82
4		HANFORD SITE; EA/SCP 5/81; DEIS/DSCP 3/82; FEIS 2/83
5		SALT DOME; EA/SCP 1/81; DEIS/DSCP 10/82
6		BEDDED SALT REGIONS[a]; EA/SCP 8/82
7		NEVADA SITE; EA/SCP 9/82
8		HARD ROCK SYSTEMS
9	PREPARE EIS	• FAR- • ENGI • DESI
10	SITE ACQUISITION	
11	TECHNOLOGY AND TESTING	FAR-FIELD SYSTEM OPERATIONAL '80; NEAR-FIELD SYSTEM OPERATIONAL '82
12	LICENSING	PREPARE AND SUBMIT SCP'S & UPDATE
13	DESIGN ENGINEERING AND CONSTRUCTION	CONCEPTUAL DESIGN
14	OPERATION	

[a]Bedded salt regions other than Los Medanos.

Figure II



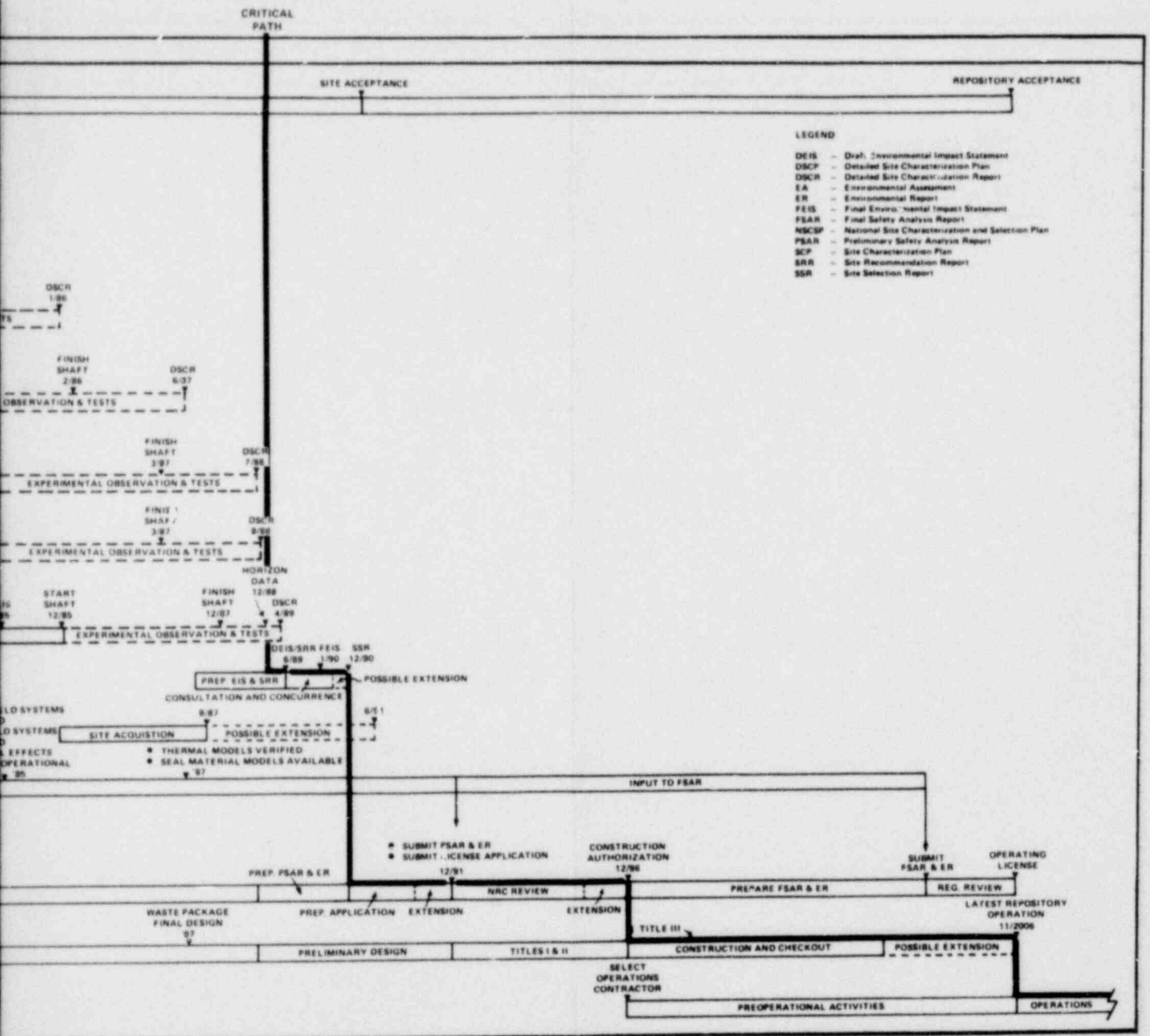
I-2. Summary Logic Network Activities Leading
to Geologic Repository Operation
(Reference Schedule)



[a]Bedded salt regions other than Los Medanos.

Figure II

(Extended dura



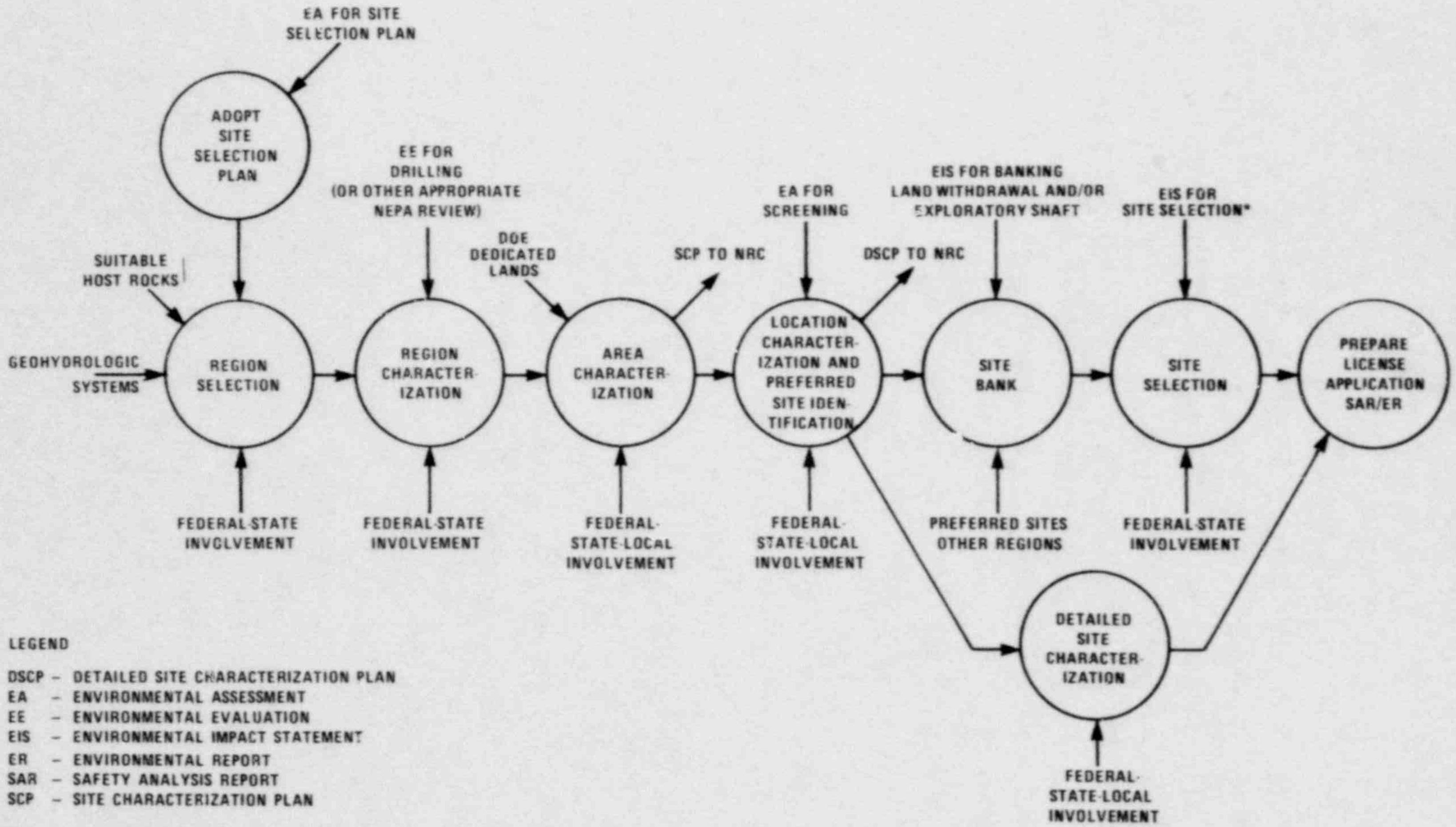
-3. Summary Logic Network Activities Leading to
Geologic Repository Operation
ion for characterization, licensing, and construction)

The data obtained in each stage of the screening process are analyzed and compared against criteria that must be satisfied for adequate performance of the total isolation system. The systems for isolation and containment of wastes comprise both natural and man-made barriers to waste movement. (The requirements for the natural systems are described in Chapter II.D, and for man-made systems in II.E.)

The screening process leads to "banking" of sites in various host rocks and geohydrologic systems. A site is banked when the participants in the siting process reach a consensus on the technical, environmental, and institutional adequacy of the site relative to established criteria, and an interest in the land has been obtained by the Department to maintain the integrity of the site through the remainder of the selection process. An Environmental Impact Statement (EIS) will be prepared for consideration in the decision to bank a site (see Figures III-2 and III-3, lines 3 through 8). When four to five sites have been banked, one will be selected for a license application to the Commission.

The NRC review of Detailed Site Characterization Plans, as proposed in a draft of 10 CFR 60 Subparts A through D (7) would be conducted in parallel with the programmatic site characterization and banking processes (see Figures III-2 and III-3, lines 3 through 8). Plans for the detailed characterization of identified sites will be submitted to the Commission for review to ensure that the proposed activities and the anticipated results will satisfy their requirements. The Commission and appropriate State and local authorities will be kept regularly informed on the progress of the Department's site characterization activities and on the progress of safety-related investigations. Environmental investigations and related decisions will be fully documented and will include public input in accordance with the National Environmental Policy Act (NEPA) (8).

The various elements that lead to final site selection are discussed below in terms of the requirements of each step and the fulfillment of these requirements. The implementation of NEPA and its relationship to site selection are more fully described in Section III.D.1.



LEGEND

DSCP – DETAILED SITE CHARACTERIZATION PLAN
EA – ENVIRONMENTAL ASSESSMENT
EE – ENVIRONMENTAL EVALUATION
EIS – ENVIRONMENTAL IMPACT STATEMENT
ER – ENVIRONMENTAL REPORT
SAR – SAFETY ANALYSIS REPORT
SCP – SITE CHARACTERIZATION PLAN

*Previous site-banking EIS's will be supplemented, as appropriate, and integrated into a site-selection EIS

Figure III-4. Site Characterization and Selection Process

III.C.1.2 Site Exploration and Characterization Process

The site exploration and characterization process involves geological and environmental studies to identify potential sites for mined geologic repositories and to obtain the technical data necessary to determine acceptability of these sites. The technical criteria to be used in the early phases are discussed in II.D.3. As the selection process narrows on more specific locations and sites, more specific criteria to be applied at these given locations and sites will be developed by the Department. Steps in the site characterization process are as follows:

1. National screening surveys.
2. Determination of regions for further study (up to several States in extent).
3. Recommendation of areas for more detailed investigation (up to 1,000 square miles).
4. Recommendation for specific locations for in-depth study (up to 30 square miles).
5. Recommendation of preferred sites for banking as candidate repository sites and associated controlled zone (nominally 10 square miles).

The status of the program in regard to the above phases in the site characterization process is discussed in II.D.5.

III.C.1.2.1 National Screening Surveys

Site searches are initiated by national screening surveys. Starting with the contiguous United States, the initial step in site exploration and characterization is to identify places that have some potential for waste isolation. These places may be regions (up to several hundred thousand square miles in area) or land areas having a particular suitability feature. National screening surveys have been structured in different ways, depending on the site suitability feature that is sought initially in process. The types of surveys used in the program are as follows:

1. A geologic approach beginning with consideration of potentially suitable host rocks and identification of regions containing these formations. Early in the program, for example, rock salt was identified as a potentially suitable host medium; thus regions in the contiguous United States containing salt domes and bedded salt formations generally suitable for repository use were identified (9). Salt domes are being studied in the Gulf Interior Region of Mississippi, Texas, and Louisiana (10). Bedded salt formations in the Permian (Texas) (11), Salina (New York, Ohio, and Michigan) (12), and Paradox Regions (Utah) (13) have either been considered or are being actively studied. The national effort has also evaluated the potential for repository development in regions containing hard rock (granite) and argillaceous (shale) formations, and recommendations on suitable regions are being developed.

2. An approach considering current land use to identify regions for further studies. Examples of this approach are the studies being conducted at the Hanford Site (14) and the Nevada Test Site (15), both of which are large tracts of land owned by the Federal Government and currently used for nuclear activities. These government reservations would be classified as "areas" in the steps in the site characterization process. Investigations of both areas were initiated to determine whether geologic and hydrologic conditions, as well as other considerations, would allow use of these dedicated lands for waste repositories. Two other Department reservations, Idaho National Engineering Laboratory (INEL) and Savannah River National Laboratory (South Carolina) have been determined to be unsuitable because of large-scale regional aquifers underlying both sites.

3. An approach based on scrutiny of successively smaller units of land based on geohydrologic conditions and then assessing whether the system of rocks within a particular geohydrologic environment has favorable repository properties. This approach provides further assurance that otherwise unexamined geologic formations having favorable repository properties will not be overlooked where they occur in a suitable geohydrologic environment. In this approach the contiguous

> United States will be examined on a province-by-province basis. The provinces are those designated by the U.S. Geological Survey. Although not all parts of the country are likely to exhibit suitable conditions, several specific areas throughout the nation are expected to prove acceptable. Of those areas that seem most promising, some will have more desirable sets of geologic conditions than others, and some may be unsuitable because of nongeologic considerations. To ensure total compatibility of provinces with siting criteria, a screening of the United States using all of these criteria (geological, hydrological, environmental, socioeconomic, political, and institutional) is expected to aid in selection of provinces to be studied later.

Whether the starting point of the site selection process is selection of regions according to rock type, land use, hydrology, or some combination of these factors, the subsequent steps in the screening process are similar. Upon completion of the national screening survey, regions are identified for further investigation. The process then continues through a series of increasingly detailed exploration activities, eventually developing detailed data on characteristics of areas, locations, and sites. These characteristics are evaluated at each level of exploration (region, area, location, and site), and geologic and environmental characterization reports are prepared. Some variations in this screening and documentation process have occurred during actual characterization studies as the program has developed. For example, screening for repository sites within the area of the NTS and at Hanford has been conducted in modified steps because of unique geohydrologic and land use features of these sites. Even though the screening process may differ from region to region, the level of investigations and final documentation will be comparable in each.

III.C.1.2.2 <u>Regional Studies</u>

Regional studies investigate the region of interest to obtain further geologic and environmental information. Studies are based primarily on a review of existing data obtained through broad literature searches.

Sources for geological data include published scientific reports and geologic maps; drilling and production records from oil, gas, and mineral exploration programs; records of earthquake occurrences and intensities; and records of water well drilling. The investigative methods used in the regional and subsequent studies are discussed in Section II.D.4.

The regional studies result in designation of the areas deemed the most suitable for further study; less promising ares are excluded. Each regional characterization effort includes preparation of a Geologic Regional Characterization Report and an Environmental Regional Characterization Report. A Regional Summary and Area Recommendation Report integrates the geologic and environmental considerations from the two reports listed above, documents the process, and provides the rationale for recommending further studies. Regional summary and area recommendation reports have been prepared which identify salt domes in the Gulf Interior Region (16), bedded salt areas in the Paradox Region (17), and in the Salina Region (18, 19) for further study.

III.C.1.2.3 Area Studies

Area studies are conducted to characterize the areas of interest designated by the regional study or designated because of their current use as DOE reservations. Environmental, socioeconomic, and geologic factors are evaluated, but within a smaller area and in greater detail than in the regional studies.

Geologic field work conducted in this phase includes drilling deep holes (several thousand feet deep) to collect rock cores for laboratory tests and to conduct down-hole geophysics tests; drilling to test the hydrologic properties of the substrata; drilling to determine the characteristics of aquifers; and conducting gravity and seismic surveys to assist in determining underlying rock structures (see II.D.4).

Environmental and socioeconomic studies are based on literature surveys of data available from local experts and institutions such as universities and local, State, and Federal agencies. The scope of area environmental studies includes a description of the hydrosphere; atmosphere; demographic, socioeconomic, and land use characteristics; and ecosystems.

The surveys of each area, conducted on non-Department lands, are documented in a Geologic Area Characterization Report and an Environmental Area Characterization Report. The summary and recommendation of locations for further study are documented in an Area Summary and Location Recommendation Report. Prior to the decision to continue further studies in the recommended locations, an Environmental Assessment will be prepared (see III.D.1).

At the completion of this stage, a Site Characterization Plan will be submitted to the Nuclear Regulatory Commission for information (see Figures III-2 and III-3, lines 3 through 8). Although this plan is not required by proposed Commission regulations at this phase of characterization, the Department intends to keep the Commission informed of its site exploration activities at an early stage and prior to identifying a preferred site in a given medium. The report will describe the site characterization process up to this phase and include plans for further detailed studies of the recommended locations. During performance of these additional studies, the Commission will be provided with periodic progress summaries for information.

III.C.1.2.4 Location Studies

Location studies further narrow the scope of the investigation to a site or sites. Geologic data gathering at this stage will include more drilling, to obtain detailed geologic and hydrologic information, and additional testing of geologic and environmental samples. Environmental studies during this phase will include complete monitoring and sampling programs at the sites to obtain specific detailed information. On-site meteorological data will be collected and compared to regional data, and physical surveys of plant and animal populations will be taken.

Information also will be obtained to document the acceptability of the sites relative to applicable Federal, State, and local regulations. These data will be documented in Geologic Location Characterization Reports and Environmental Location Characterization Reports. A Location Summary and Site Recommendation Report, integrating the geologic and environmental considerations, documents the process and provides the rationale for recommending sites for further detailed studies.

This phase ends with the banking of a preferred site, as discussed in III.C.1.1. The banked site will be one of four to five candidate sites to be considered for the first repository license application. An Environmental Impact Statement will be prepared for consideration in the decision to bank a preferred site in a particular medium or system.

Based on current plans, it is expected that Final Environmental Impact Statements will be completed for the Los Medanos Site by November 1981, for a potential repository site within the Hanford Site by February 1983, for salt domes by July 1983, for bedded salt by September 1984, for a site within the Nevada Test Site by November 1984, and for a hard rock system by June 1985 (see Figure III-2, lines 3 through 8).

At the completion of location studies, a Detailed Site Characterization Plan will be submitted to the Commission for review (7). The report will fulfill all the requirements, including description of the characterization process up to this phase and a detailed description of the studies to be performed during the following phase of characterization of the preferred site. The purpose of this NRC review is to ensure that the site characterization studies, when complete, will have acquired the necessary information for licensing without compromising the integrity of the site. Periodic progress summaries will be submitted for review.

III.C.1.3 Detailed Site Characterization

A site for a repository includes nominally about 2,000 acres, which might be used for actual subsurface development; the size may vary from about 1,200 acres to 5,000 acres, however, depending on several factors such as the extent of the subsurface isolation system. In addition the site would include a controlled zone in which the Department would restrict future activities.

The purpose of detailed site characterization is to collect all additional data that would be necessary if a license application were submitted for that site. Data gathering methods will include more extensive drilling to obtain geologic and hydrologic information, on-site and laboratory testing of rock and water samples, and fine-mesh geophysical surveys. The underlying rock structure will be characterized in sufficient detail to establish engineering and design envelopes and to confirm safety assessments and construction feasibility. Depending on the ability to characterize fully the condition of the site, it may be necessary to proceed with an exploratory shaft and at-depth characterization activities at this time.

The majority of data on environmental impacts will be collected prior to the detailed site characterization phase and evaluated in the EIS prepared for the banking decision. Any environmental information gathered in the detailed site characterization phase will be incorporated into the environmental impact statement prepared for the selection of the site for the first repository (see III.D.1). The product of this phase will be a Detailed Site Characterization Report.

Detailed Site Characterization Reports based upon observations from the surface only would be complete for the Los Medanos Site by August 1982, salt domes by September 1983, at the Hanford Site by March 1984, for bedded salt by April 1985, at the Nevada Test Site by June 1985, and for hard rock systems by December 1985. These dates are based on acquisition of the data required for adequate characterization of the repository horizon by surface exploration techniques such as boreholes and geophysical tests.

If such data must be obtained by sinking shafts and exploring at the repository depth prior to submitting an application for construction authorization, these dates could be delayed by about 40 months, as shown in Figure III-3. In this case, the enviromental impacts of sinking an exploratory shaft would also be discusssed in the EIS. Any construction work on such a shaft would take place only after issuance of the Final EIS. In the case of the Hanford Site and the Los Medanos Site, such work could begin after issuance of the Final EIS. Figure III-3, however, shows additional time for completion of and acquisition activities at other sites prior to actual shaft construction. In the case of Nevada, 4 months are allowed for possible procedures concerning lands on or immediately adjacent to the Nevada Test Site. In the case of other formations, a total of 6 months is allowed for completion of acquisition procedures. Though contracting and contractor mobilization activities for a shaft could be accomplished in parallel with the reference scheduled activities, times of up to 24 months have been allowed in the schedule for shaft construction and installation of experimental equipment. An additional 16 months have been allowed for repository horizon experimental observations, tests, and data analysis.

The 24-month estimate for shaft construction assumes blasting and excavating a shaft with a diameter of 18 feet. A shaft of this size would be required if extensive development of drifts for examination of the subsurface geology were contemplated. More detailed analysis of possible exploratory shafts to support subsurface investigations has been completed in the case of the Los Medanos Site and the Hanford Site. Conditions at Los Medanos are such that it is estimated that an exploratory shaft could be completed in 10 months followed by a 14 month period for subsurface investigations. The basalt waste isolation program has estimated that subsurface investigations at Hanford might require underground excavation of no more than 500 feet which could be supported by a 9-foot diameter shaft. Such a shaft could be completed in 16 months.

In cases where shafts are found to be required for exploration, the dates for issuance of Detailed Site Characterization Reports would be for the Los Medanos Site November 1983, the Hanford Site January 1986, salt domes June 1987, bedded salt July 1988, Nevada Test Site September 1988, and hard rock April 1989.

III.C.1.4 Site Selection Process

After four to five sites have been found suitable and have been banked, a particular site will be selected for a license application to the Commission. The formal site selection process can be initiated after several candidate sites in alternate media and systems have been banked. As shown in Figure III-2, the banking of the hard rock system site is estimated to take place in June 1985.

Meeting this date would require the ability to acquire all necessary characterization data for license application by surface exploration methods. If it were deemed necessary, for technical reasons, to explore at the repository depth (horizon) by constructing shafts and drifts, then the site selection process would not be initiated until confirmatory repository horizon data were obtained and analyzed. In this case, initiation of the selection process would follow availability of horizon data for the hard rock system in December 1988, as shown in Figure III-3, line 8.

The process for final site selection will include a comparison of environmental factors and technical aspects of the banked sites plus the legal, political, and institutional considerations on a national scale. The environmental factors will be compared in the EIS prepared for the candidate site selection (see III.D.1). The comparison of all factors will be documented in the Site Recommendation Report (SRR) (see Figures III-2 and III-3, line 9). A 6-month period is allowed for preparation of the SRR following the banking of the latest candidate site and the initiation of the site selection process.

The Site Recommendation Report will be reviewed by appropriate Federal and State agencies which have responsibility in the siting process. In addition, the report will be available for public review, as prescribed in the President's message of 12 February 1980 (1). When Federal and State agencies have reached a consensus on the SRR, it will be updated and issued as a Site Selection Report.

The process of site characterization and banking of multiple candidate sites, coupled with parallel review by the Commission, State and local governments, and public participants, will result in a thoroughly reviewed and accepted site. It is estimated that the site selection decision can be achieved within 15 to 18 months following issuance of a Site Recommendation Report.

The schedules shown in these figures relate to the selection of only the first site for a repository. Continued characterization activities being carried on in geohydrologic provinces and those regions identified by the planned national screening will provide additional candidate sites that will be banked for consideration for the second and subsequent repositories.

III.C.1.5 Summary

The selection and banking of candidate sites will be based on a systematic process that considers all applicable factors and includes full public participation. Regional and area characterizations are now under way in various geologic media, including dome salt and bedded salt formations, basalt flows, and volcanic tuff. Efforts have also been initiated to identify regions in other media, and a systematic national screening for other geohydrologic systems is planned.

The Department plans to identify at least four sites with diverse rock types by mid-1985. The Department's approach ensures that consideration will be given to regulatory and environmental factors, the necessity of achieving public acceptance, and the need to meet site qualification criteria.

III.C.2 State Involvement in Selection of Repository Site

III.C.2.1 Introduction

The involvement of States in the radioactive waste management program was addressed by the President in his statement of 12 February 1980 (1) as follows:

> First, my Administration is committed to providing an effective role for State and local governments in the development and implementation of our nuclear waste management program. I am therefore taking the following actions:
>
> o By Executive Order, I am establishing a State Planning Council which will strengthen our intergovernmental relationships and help fulfill our joint responsibility to protect public health and safety in radioactive waste matters. I have asked Governor Riley of South Carolina to serve as Chairman of the Council. The Council will have a total of 19 members: 15 who are Governors or other elected officials, and 4 from the Executive departments and agencies. It will advise

the Executive Branch and work with the Congress to address radioactive waste management issues, such as planning and siting, construction, and operation of facilities. I will submit legislation during this session to make the Council permanent.

- In the past, States have not played an adequate part in the waste management planning process--for example, in the evaluation and location of potential waste disposal sites. The States need better access to information and expanded opportunity to guide waste management planning. Our relationship with the States will be based on the principle of consultation and concurrence in the siting of high level waste repositories. Under the framework of consultation and concurrence, a host State will have a continuing role in Federal decisionmaking on the siting, design and construction of a high level waste repository. State consultation and concurrence, however, will lead to an acceptable solution to our waste disposal problem only if all the States participate as partners in the program I am putting forth. The safe disposal of radioactive waste, defense and commercial, is a national, not a Federal, responsibility.

- I am directing the Secretary of Energy to provide financial and technical assistance to State and other jurisdictions to facilitate the full participation of State and local government in review and licensing proceedings.

State involvement in site selection is an important aspect of the radioactive waste disposal program (20). Individual States are now and will continue to be involved in the site selection process, starting with the initial step of regional explorations and studies. The "consultation and concurrence" approach follows the recommendations of the President and the Interagency Review Group (21) and integrates with requirements of proposed 10 CFR 60, 10 CFR 51, and the review processes of the National Environmental Policy Act (8) as stipulated by the Council on Environmental Quality (22).

Early consultation with State and local officials is sought from the time any portion of the State is specifically considered. This consultation will be sought by the Department, in briefings, transmittal of supporting documents, formal letters, and public information meetings, and will allow for continuous interaction between State and local representatives. The Department will exchange ideas with the State and respond to and incorporate the State's concerns into the program.

III.C.2.2 Consultation Process Under Way Within States

The Department is developing an information program to inform State and local officials of investigative activities and repository development and to involve such officials in decisionmaking for site selection and repository licensing.

Proposed Commission policies and regulations for review and licensing of high-level waste repositories call for early notification of State officials and other interested parties and opportunities for State participation in the NRC review process. Similarly, previous Commission studies on this topic (20, 23) have focused on the need for early State involvement.

The consultation and concurrence process is one of several avenues by which societal concerns about the disposal of nuclear waste are being evaluated. The Department has taken several steps to encourage advice from independent groups; it has sponsored a national workshop on consultation and concurrence (24), established independent program and technical review committees (composed of persons in national leadership roles in various fields), consulted with other Federal departments and agencies, and assisted in establishing the State Planning Council recommended by the Interagency Review Group. The Council was established by the President on 12 February 1980, with the purpose of providing State and local perspectives in the development and recommendation of a comprehensive nuclear waste management plan. The Council, which is chaired by Governor Riley of South Carolina, held its first organizational meeting on 24 February 1980. A staff to serve the Council is being organized.

The consultation process already has been initiated in New Mexico, Utah, Louisiana, Mississippi, Nevada, Texas, Michigan, and Washington. Public information meetings have been held, and information briefings with State and local officials, formal letters, and supporting documents have been provided. Specific activities are described below.

III.C.2.2.1 National Screening Survey

The key consultation and concurrence aspect of the national screening phase is the establishment of initial communication and interaction between Federal and State officials. The Department actively seeks State participation in the work to characterize various geohydrologic systems. In the screening of the geohydrologic provinces, for example, the U.S. Geological Survey will seek participation of all State geologists within a province. After completion of the national screening phase, officials of participating States will receive reports summarizing results of studies and will be asked for advice concerning plans for the next step in the site characterization process.

III.C.2.2.2 Regional Studies

States in which regional studies are planned are notified by the Department. Federal officials work with the Governor, the Legislature, and any special committee during the exploration and characterization process.

The Department also works with specific State officials and technical agencies and seeks their views and advice on exploration plans, so that as studies progress from regions to locations, appropriate understandings and agreements are in place. Each of the Department's regional offices (located in the 10 Federal regions) is directed by a Regional Representative, who seeks to establish continuous liaison between the Department and the State governments of that region. The Department's field operations offices also interact with State governments, as appropriate, in carrying out their program responsibilities. This procedure continues through the studies during the area, location, and site characterization phases of the program.

An example of such interaction is recent consideration by officials in the State of Michigan that regional studies including the State should be completed. The evaluation of the Salina salt formations in the State of Michigan was discontinued in 1977 as the result of concerns of State authorities which were not satisfied. However, a reassessment of the impact of a State law prohibiting radioactive waste storage was initiated by Governor

Milliken in 1979. State officials were provided information on the NWTS Program at a meeting in Lansing, Michigan, on 12 September 1979. The Governor has appointed a special committee of representatives from the State administration, the legislature, and the private sector to provide the legislature with information for use in its assessment of nuclear waste activities for the State of Michigan. On 14 February 1980, representatives from the Department and ONWI again met in Lansing with the special committee. The committee then voted to recommend to Governor Milliken that Michigan permit the Department to complete the regional studies of the Salina salt basin. The committee also recommended that a working group, consisting of Michigan scientists be established to work with the Department in compiling and reviewing the regional Michigan geologic and environmental data.

III.C.2.2.3 Area Studies

Interaction among the Department and State and local officials becomes more frequent during area studies; interest is more focused on the specific parts of a State that are under study, and residents and local officials become more directly involved. Exploratory work involves activities such as drilling, sampling, and installation of equipment, and necessitates obtaining permits, licenses, and land access agreements. Contacts with State, county, and city officials, and the general public increase commensurately.

Specific examples of such interactions are meetings held in March, May, and November of 1979 with the Mississippi Governor's Select Committee on Nuclear Energy and Nuclear Waste Repository, and briefing of the Mississippi Wildlife Management Commission; and, in July and August of 1979, briefings of the Governor of Utah, the Utah State Energy Council, the Joint Interim Energy Committee of the Utah Legislature, and the Commissioners of Grand and San Juan Counties in Utah.

Funded through grants from the Department, public information and program oversight activities are being conducted by the Louisiana Office of Conservation, the Mississippi Energy Office, and the Texas Energy and Natural Resources Advisory Council. State involvement also has come through research contracts with institutes and faculty members at universities.

Louisiana State University, the University of Southern Mississippi, and the University of Texas are some of the institutions currently thus engaged.

A joint group involving State and Department officials has been established in the State of Washington. At the suggestion of Governor Ray in April 1979, the Department and the State of Washington formed a Basalt Working Group to strengthen the involvement of that State in the Basalt Waste Investigation Program. State officials include representatives of the Governor's office and legislative leaders. Meetings to discuss BWIP status and development are scheduled on a quarterly basis.

In Nevada, informal interactions between the Department and the State have been established. A mechanism to formalize these interactions will be developed in the near future.

Congressional delegations will be kept informed of the status of the programs. Public meetings will be held in localities near sites where investigations are being conducted, and local media will be kept up to date on all developments. The advice of State and local technical experts will be solicited at appropriate points in the investigation. All of these activities will continue through the location and site studies.

III.C.2.2.4 Location and Detailed Site Characterization Studies

At this stage, more direct contact between the Department and local officials and citizens will be established while maintaining communication channels with State government officials. Local information will be provided through many types of communication media. A Department representative may be assigned to specific locations to provide a source of information to local communities.

An example of interactions with a local community is associated with investigations of the Los Medanos Site in New Mexico that have previously been undertaken as part of the Waste Isolation Pilot Plant (WIPP) project (this was part of the defense waste management program not included in the NWTS Program). Interactions with State and local authorities in New Mexico have been conducted in a manner similar to that described here. In particular, the Department has provided funds to the State of New Mexico to establish an independent review body to examine technical issues associated with the proposed site. The Government of the State of New Mexico has established the Environmental Evaluation Group as this independent body. The Department has provided funding for this body in an agreement with the State under which complete independence and access to all data is specified.

The President has decided that the WIPP project should be canceled and that the site should be evaluated along with other candidate sites in the NWTS Program. As examination of the Los Medanos site is integrated into the NWTS Program, consultation with State and local officials and funding of the Environmental Evaluation Group will continue.

III.C.2.3 Relevance to Siting Decision

Previous history and testimony, exemplified by the Department and Commission workshops on radioactive waste management (20, 23, 24), have shown that close interaction among the Federal Government, State, and local officials will be necessary to establish a radioactive waste disposal site. A continuing interest has been expressed by many States in the Federal Government's waste disposal program as evidenced by the ongoing activities and legislative actions taken in several States (23). Consideration must be and is being given to political and institutional issues because they can affect the timing and viability of site selection.

The consultation and concurrence process will provide for consideration of State and local interests in repository siting. It will augment rather than replace other important regulatory processes and reviews that a repository site must undergo. The process to be followed by both Federal and State officials, coupled with a mutual recognition of the need to resolve waste management issues in the national interest, will enable repository site selection to be achieved in a timely fashion.

III.C.2.4 Summary

In summary, a process is being developed that provides for cooperative Federal, State, and local government decisionmaking with respect to identifying candidate sites and selecting one for a license application.

This approach involves the Department, State, and local officials in all phases of the site selection process and allows for continuous response to State concerns. In addition, public hearings and reviews, which are part of the regulatory licensing and permit procedures, will provide opportunities for resolution of conflicts.

The Department's program seeks to take into account State and local needs and concerns. It is anticipated that open and full sharing of information will enable early identification of any impacts on schedule, so the program then can be adjusted accordingly. The schedules presented in Figures III-2 and III-3 are based on successful State and public involvement in the site characterization and selection process, thus allowing field exploration activities to continue, from region, to area, to location phase, without undue delay. It is possible that unanticipated or unresolved issues of concern at the State or local level could cause prolonged perturbations in the schedule. However, as the President stated in his message on waste management of 12 February 1980, ". . . all must understand that this problem will be with us for many years. We must proceed steadily and with determination to resolve the remaining technical issues while ensuring full public participation and maintaining the full cooperation of all levels of government."

III.C.3 Licensing of Repositories

The Nuclear Regulatory Commission has the statutory authority to license facilities used primarily for the receipt and storage of high-level radioactive wastes resulting from activities licensed under the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974 (25). The licensing process, therefore, is an integral factor in the Department's plans to develop mined geologic disposal. Two issues regarding that process have an immediate impact on the Department's ability to achieve a licensed disposal system within the schedule presented. First, both the Commission and the Environmental Protection Agency are in the process of developing regulatory requirements for mined geologic disposal. Second, the Commission will be asked by the Department to license a first-of-a-kind facility. Therefore, the procedures that are utilized will have an impact. The Department's efforts to address these issues are discussed below.

III.C.3.1 Establishment of Regulatory Requirements for Mined Geologic Disposal

The Department will likely be required to obtain the opinions of the Commission staff in regard to field exploration activities for mined geologic disposal, so that the integrity of the site is not adversely affected (7). Accordingly, the availability of regulatory procedures and requirements could have an impact on the schedule of a disposal system. This section examines the potential impact and the steps undertaken by the Department in this regard.

On 17 November 1978, the Commission published a proposed policy statement regarding establishment of procedures for licensing geologic high-level waste repositories (26). The Department has structured its program to comply with the spirit and intent of the published procedures, although some of the details of implementation are still under development. Based on testimony to the Congress by the Commission (27) and information provided by Commission staff at a public meeting on 4 October 1979, a final rule relating to the procedural aspects of licensing is anticipated during calendar year 1980. The proposed rule was published for comment on 6 December 1979 (7). Therefore, the Department is aware of the likely procedural requirements of the Commission that could have a major impact on cost and schedule, and has considered them in preparing program plans.

The Commission's technical requirements also can directly affect cost and schedule. An independent technical assessment and review by the Commission will be based on predetermined NRC criteria and requirements. The ability of the Commission to reach a satisfactory conclusion regarding the safety aspects of an application will be directly affected by the manner and degree of demonstrated conformance with such criteria and requirements. Although the Department is using conservative criteria, approaches, and methods, there is a need to ensure that this approach will be compatible with that required by the Commission and amenable to timely Commission review. In this regard the Department provides continuing reports of its R&D programs to both the Commission and the Environmental Protection Agency for the purpose of informing both agencies of the current status of the Department's effort.

Based on information presented at the 4 October 1979 Commission staff briefing for the Department, the Commission expects to publish the proposed technical criteria in mid-1980 and the final rule in mid-1981.

The Environmental Protection Agency is responsible for developing standards applicable to all Federal radioactive waste management programs; these standards will be implemented in NRC regulations. EPA has published for public review the initial formulations of their standards (28). Three public workshops to discuss the contents were held in February and April 1977 and in March of 1978. The Department has participated in these discussions and is sharing data with EPA for its use in formulating standards. Publication of standards for high-level waste disposal for public review and comment is anticipated for the spring of 1980.

In light of the schedules for NRC and EPA regulations presented above and of available information, the Department's current program is sufficiently flexible that the projected schedule will not be greatly altered as a result of codification of their regulatory requirements. It is currently understood that adequate regulatory guidance will be available in sufficient time so that the program schedule can be accomplished. For example, even though some issues, such as the need for at-depth exploration of multiple banked sites, are still being discussed, the DOE program is sufficiently flexible to accommodate the outcome of these discussions.

On the basis of comments received from NRC and EPA on documents such as the draft environmental impact statement of commercial waste management, there is agreement with the geologic disposal planning strategy being pursued by the Department. Specifically, NRC staff stated (29):

> We agree that a repository should be developed and tested as soon as possible.

Similarly, EPA has stated (30):

> We agree with the Department that the option selected for implementation appears to be the best of those considered ... . It is also unlikely that there would be any viable alternative available in the near future. For this reason we believe DOE's program should be vigorously pursued.

Both the Commission and EPA have been proceeding with the development of standards and criteria for geologic repositories. The specific determination regarding the degree of conformance of Department-developed requirements with those eventually promulgated by the Commission and EPA could be delayed until the start of the licensing review. In such an event, the Department believes that any newly promulgated NRC or EPA requirements can be met within the time intervals planned for in the licensing process.

III.C.3.2 The Licensing Procedures

The preceding discussion described the program's ability to proceed in light of developing regulations. The following discussion presents the impacts of the anticipated NRC licensing process on the repository development schedule.

The Commission published a proposed rule regarding the NRC licensing procedures in the Federal Register on 6 December 1979 (7). This document identified four parts to the licensing process: (i) Review of the Department Site Selection, (ii) Construction Authorization Application Review, (iii) Repository Licensing and License Amendment, and (iv) Review of Repository Decommissioning. At a briefing held on 4 October 1979, NRC staff outlined to the Department its current thinking on implementation of the process.

Consistent with the intent of proposed NRC regulations, the review of site selection would begin with the Department filing a Detailed Site Characterization Plan (referred to as Site Characterization Report in draft 10 CFR 60.11) with the Commission at the time that a preferred site is designated for detailed characterization studies. The plan would describe (i) the screening process that the Department used to identify the site, (ii) the characterization procedures to be used in studying the site, and (iii) a discussion of status of identifying other sites for characterization. The Commission has announced that it expects the Department to characterize a minimum of three sites representing a minimum of two geologic media before one is selected for a Construction Authorization application (31). As discussed in Section III.C.1, the Department plans to have at least four sites in several rock types identified by mid-1985.

Before a decision on Construction Authorization is made, an affirmative decision regarding site suitability must be made. To enable such decisionmaking, the site has to be sufficiently characterized. The degree of investigative activity necessary to adequately characterize a site has not yet been completely determined. Resolution of this item requires that the Commission identify the information it considers necessary for a finding at the construction authorization stage. Depending on the requirements for site-specific information, characterization could require as much as an additional 40 months beyond currently planned activities, and would be determined by the type of measurement techniques that would be employed. This additional time would involve construction of an early shaft and exploration at the repository horizon prior to application for a construction authorization.

The Commission staff indicated at the meeting on 4 October 1979 that it estimates that 48 months will be required to review a construction application and to hold public hearings (see Figures III-2 and III-3, line 12). The early familiarity with the project that the Commission staff gains by reviewing the Detailed Site Characterization Plan, and a working familiarity with the R&D projects, may permit completion of review in a shorter interval. Nevertheless, the Department has used an elapsed time from filing of the application to a favorable construction authorization decision of 48 months, as suggested by the NRC staff, in order to estimate the total time needed to develop a repository. An additional 12 months has been allowed in the extended schedule for unanticipated delays that may occur. These assumptions are consistent with staff suggestions that time intervals of 10 to 12 years from preliminary design to a decision on operating approval should be used in establishing repository availability dates.

The next step in the licensing process would be application for a license to receive and possess the wastes. This activity can be conducted in parallel with repository construction and should not be on the critical path (see Figures III-2 and III-3, lines 12 and 13).

The remaining phases of licensing of a repository are amendments to allow nonretrievable storage, decommissioning, and termination of the license. Since all these activities would follow receipt of the waste into the repository, they will not affect the date at which a repository will be available.

III.C.3.3 Summary

The existing knowledge of licensing requirements, as obtained from draft and proposed rules and communication with regulatory agencies, has allowed the bounding of the effects licensing activities will have on the schedule. Based on the existing information, estimates can be made of the time which will be required for licensing. The attainment of such milestones will depend on the final rule promulgated by the Commission and on the manner in which the Commission will process future license applications.

III.D FACTORS INFLUENCING SCHEDULE FOR MINED GEOLOGIC DISPOSAL

In addition to the major decisions discussed in Chapter III.C, several significant factors also can influence the timing and schedule of a repository. This chapter discusses each factor, its relevance to the schedule, programs under way to resolve questions or choose preferred methods, and the associated schedules.

III.D.1 Implementation of National Environmental Policy Act

III.D.1.1 Introduction

The National Environmental Policy Act of 1969, (NEPA) (8), as implemented by the regulations of the Council on Environmental Quality (CEQ) (22) and the Department's guidelines (32), requires that potential adverse environmental consequences be considered in Department planning and decision-making. In managing the National Waste Terminal Storage (NWTS) Program, the Department will undertake actions having potential environmental consequences. The potential environmental effects of these actions and their significance vary. Actions range from decisions on an overall strategy for waste disposal (involving a major resource commitment which ultimately may have a spectrum of potential environmental effects specific to that strategy) to the selection of specific sites and facilities for waste disposal purposes. Other actions include the conduct of research (data gathering and analysis) which may have little environmental effect but which may have important technological, cost, and time implications on long-term waste disposal.

Using the Department guidelines and the CEQ regulations, the Department has developed a NEPA implementation plan for the deep geologic disposal interim strategy which is integrated with the overall Department planning and decisionmaking framework discussed in Chapter III.A. Figures III-2 and III-3 graphically demonstrate the integration of the NEPA plan and the overall decisionmaking process.

The Department's NEPA implementation plan for the NWTS program is based on the "tiered" approach, which is designed to eliminate repetitive discussions of the same issues and to focus on the actual issues ripe for decision at each level of environmental review. This approach allows coverage of general matters in broad Environmental Impact Statements (EIS's) with subsequent narrower EIS's or Environmental Assessments (EA's) incorporating by reference the general discussions and concentrating solely on the issues specific to the subsequent decision.

The NEPA implementation plan identifies the major decision points in the program to assure that appropriate environmental documentation is completed prior to each such decision and prior to the conduct of activities that may cause an adverse environmental impact or limit the choice of reasonable alternatives. The first major decision process in the NWTS program is selection of a program strategy for disposal of nuclear waste. As will be discussed more fully below, the mined geologic option is now being pursued as the interim planning strategy.

The second major decision process is that involving the selection of sites for the disposal of nuclear waste assuming the mined geologic option. The process for selecting sites for geologic disposal is described in Section III.C.1. The major decision points in the site selection process are:

1. Adoption of a National Site Selection and Characterization Plan including the National screening for potential regions and selection of areas (approximately 1,000 square miles) for further study.

2. Identification of locations (10 to 30 square miles) for in-depth study.

3. Selection of a preferred site(s) for banking, including the possible development of an early shaft.

4. Acquiring an interest in land, including action to protect potential sites from other uses.

5. Selection of a candidate site to propose to NRC for licensing as the first repository.

At each of these decision points, the Department will consider the appropriate NEPA documentation. While the appropriate NEPA documentation is being prepared for the various decision points, program activities, including site characterization activities, that have been analyzed in previous NEPA documents may continue. In addition, further site characterization activities may continue, if it is clear, based on the Department's review, that they do not (i) have an adverse environmental impact or (ii) limit the choice of reasonable alternatives (33). These activities could include environmental studies, routine geophysical studies, shallow drilling, and borehole drilling.

III.D.1.2 The Department's National Environmental Policy Act Implementation Plan

III.D.1.2.1 Program Strategy

The environmental effects of selecting a program strategy, including the selection of a preferred technical concept for waste disposal, are addressed in the Draft EIS on Management of Commercially Generated Radioactive Waste (34), issued in April 1979 for public comment. Ten technical concepts, including mined geologic disposal, are analyzed in the Draft EIS. The substantive issues raised through the public comment process have been carefully reviewed and are being addressed in the Final EIS which is scheduled for issuance by October 1980.

Pending the completion of the Final EIS, the Department is implementing the President's interim planning strategy which is focused on the use of mined geologic repositories (1).

III.D.1.2.2 Site Selection Process

III.D.1.2.2.1 National Site Characterization and Selection Plan

The Department proposes to adopt formally the current NWTS Site Characterization and Selection Plan as the comprehensive National Site Characterization and Selection Plan. The current plan incorporates the process described in III.C.1, and will be followed pending adoption of a formal plan. An EA is being prepared as input to the decision on whether to adopt or modify this plan.

The proposed plan includes:

1. The methodology for identifying geographic regions for site studies.
2. The methodology and criteria for screening these regions for areas, locations, and candidate sites to be studied in detail.

The environmental impacts of the methodology and criteria in the proposed plan and their reasonable alternatives will be assessed. In addition, the selection of areas for further study and the anticipated range of site characterization activities, including the environmental impacts of typical surface and subsurface activities in several environmental settings, will be analyzed. Similarly, the criteria proposed to be used to qualify and disqualify sites will be discussed.

It is believed that an EA, and not an EIS, is the appropriate level of NEPA review, since it is unclear that the decision will have environmental significance. However, upon completion of the EA, a decision will be made regarding the need to prepare an EIS. The Department will consider the results of the NEPA review prior to deciding whether to adopt or modify the proposed plan. The adopted site characterization process will be repeated in diverse geologic environments and different host media until four to five sites have been qualified.

III.D.1.2.2.2 Identification of Locations

Following completion of Area Studies for a particular region, in accordance with the National Plan, an EA will be prepared as input for a decision to narrow the investigations to a limited number of locations. The site selection process to date will be described, and the environmental factors pertinent to the proposal to limit more comprehensive exploratory activities to the preferred locations will be analyzed. A comparison of environmental factors for preferred and alternate locations, based on data commensurate with the level of site-specific information available, will be provided and the environmental impacts of the range of potential exploratory activities anticipated in the location studies will be considered.

Here too, it is believed that an EA is the appropriate level of NEPA review, since it is unclear that this decision will have environmental significance. However upon completion of the EA, a decision will be made regarding the need to prepare and EIS.

III.D.1.2.2.3 Identifying Preferred Sites for Banking/Early Shaft

At the conclusion of the location studies, the Department will propose one or more of the sites in a location as a preferred site to be banked (see III.C.1). Because a banked site ultimately may become the location of a repository, it is appropriate to prepare an EIS prior to the decision to bank the preferred site(s). This EIS also would provide input to a decision to acquire an interest in the site(s), if necessary, in order to maintain the integrity of the site through the site selection process.

Using a general conceptual design for the appropriate media (a site-specific design will not be developed until after the candidate site is selected), the EIS will evaluate the potential environmental impacts of (i) a conceptual repository at the alternate sites within the region (ii) the detailed site characterization activities which may be required at each of the alternate site(s), including the possible construction of an early shaft, if required.

Although the general conceptual design will not be site-specific, it will be in an advanced stage of development relative to the medium in which the potential candidate sites are located. This will allow adequate analysis of the potential environmental impacts associated with a conceptual repository at each of the alternative sites. In addition, the interaction of waste package options with the geologic medium will be assessed in each site banking EIS.

III.D.1.2.2.4 Site Selection

Following the banking of sites in several media, a site will be selected for a license application for the first repository. The EIS's previously prepared for site banking will be supplemented, as appropriate, in an integrated EIS, which will provide a comparative environmental analysis of the alternative sites. This EIS will incorporate by reference the site-banking EIS's and include any significant new information obtained since the preparation of the earlier EIS's. The site-selection EIS also will serve as input to the environmental report submitted to NRC with the license application.

III.D.1.2.3 Land Acquisition

After a site selection decision, the Department may take steps to permanently acquire the site. The site banking EIS's, as supplemented in the site selection EIS, will be used as input to the land acquisition decision.

III.D.1.3 Schedule and Updates

As graphically demonstrated in Figure III-2, all NEPA documentation will be coordinated with the preparation of technical reports and provide timely input into the Department's decisionmaking. The Department's NEPA implementation plan for the repository program will be updated periodically as necessary, to reflect current policy.

III.D.1.4 Summary

A management approach which integrates NEPA requirements into overall program planning and decisionmaking has been developed. The management approach is structured to ensure that the environmental impacts of all reasonable alternatives will receive meaningful consideration at each stage of the decisionmaking process. The Department believes that this management approach will result in the timely attainment of a safe and environmentally acceptable operating waste disposal system.

III.D.2 Cooperation of Multiple Federal Agencies

The Presidential statement of 12 February 1980 emphasizes the commitment to provide for safe disposal of radioactive wastes with support from all agencies within the Administration, as follows:

> My objective is to establish a comprehensive program for the management for all types of radioactive wastes. My policies and programs establish mechanisms to ensure that elected officials and the public fully participate in waste decisions, and direct Federal departments and agencies to implement a waste management strategy which is safe, technically sound, conservative, and open to continuous public review. This approach will help ensure that we will reach our objective--the safe storage and disposal of all forms of nuclear waste.

In addition to emphasizing the interactions among agencies with expertise and responsibility for elements of a waste management program, the Administration has placed a high priority on the program, as evidenced by the Interagency Review Group (IRG) effort (21). In its role as lead agency for the management and disposal of radioactive wastes, the Department is preparing, with cooperation of other cognizant Federal agencies, a detailed National Plan for Nuclear Waste Management to implement the President's policy guidelines and other IRG recommendations. The Department's program has been reviewed at the highest levels of government and by a variety of agencies and organizations over the last 2 years. The program content now includes many activities specifically recommended by the IRG so that other agencies will support the Department activities where required. The ability to draw on the resources of such organizations and to obtain meaningful comments and direction will enhance the Department's ability to meet major milestones.

In accordance with the President's direction, the Department has established an Interagency Working Committee on Radioactive Waste Management. This committee is chaired by the Department's Deputy Assistant Secretary for Nuclear Waste Management and is composed of officials from other agencies such as the Department of the Interior, the Department of Transportation, the Environmental Protection Agency, and the Nuclear Regulatory Commission. The goal of this committee is to ensure that the President's waste management policy is properly implemented. The functions include interagency communications and coordination on technical and nontechnical matters. Specific arrangements for interagency cooperation are described in the following sections.

III.D.2.1 U.S. Department of the Interior

III.D.2.1.1 U.S. Geological Survey

Coordination with the U.S. Geological Survey (USGS) of the Department of the Interior occurs through a variety of mechanisms such as periodic planning meetings, reports of technical progress, and information exchanges on the parallel programs of the Department and the U.S. Geological

Survey. A formal Memorandum of Understanding will shortly be developed between the Department of the Interior and the Department of Energy covering this cooperation with both the USGS and the Bureau of Land Management (BLM). This Memorandum of Understanding will include:

1. Procedures for reserving potential repository sites on public land.

2. Procedures for collaboration on the summaries of the status of knowledge relevant to disposal of high level and transuranic wastes.

3. Procedures for collaboration on the site qualification and Earth Sciences components of the Nuclear Waste Management Plan, and for cooperation in the areas of research and development and site qualification activities, including mechanisms for transfer of funds as appropriate.

4. Procedures for the Department of the Interior to assist and advise the Department of Energy in the conduct of studies relevant to rock mechanics of repositories, in reviewing repository designs, and in monitoring repository construction activities.

5. Implementation of consultation and concurrence in relations with States and liaison with the State Planning Council.

6. NEPA implementation planning.

7. Other relevant matters as mutually agreed.

The unique and germane experience of the USGS is being utilized to support a variety of technical activities, as outlined below.

III.D.2.1.1.1 Earth Sciences Technical Plan

The USGS is participating with the Department in development of an Earth Sciences Technical Plan (55) to define the technical efforts required for successful mined geologic waste disposal. This plan describes technical efforts required in such areas as site identification and characterization, rock mechanics, repository sealing, waste/media interactions, and repository performance assessment. It will be used to assist in the development of detailed planning of activities in the NWTS Program.

III.D.2.1.1.2 Evaluation of Potential Geological Environments

The USGS is involved in geological/hydrological characterization activities that are funded through Interagency Agreements (35) as part of its support of the NWTS Program. The USGS has a large reservoir of germane information as well as capability to develop additional technical data. These capabilities and programs have been applied to support the Department's program, as described in the Earth Sciences Technical Plan and in numerous USGS open-file reports. For example, the USGS is coordinating and conducting studies of cored geologic material, geophysical surveys to characterize areas of interest, and remote sensing studies, within the Paradox Basin; is compiling geohydrologic maps for the South Central Mississippi Salt Dome Basin and the North Louisiana Salt Dome Basin; and, in cooperation with the Department, has planned to perform geologic/hydrological evaluations of the United States. The USGS-designated provinces are shown in Figure III-5. These studies will be conducted by funds directly allocated to the USGS.

The USGS also is conducting the site exploration and site characterization activities in Nevada, as well as the regional hydrologic studies for the Basalt Waste Isolation Project.

III.D.2.1.1.3 Technology Development

USGS expertise is being utilized in the conduct of certain technology studies defined in the Earth Sciences Technical Plan. For example, the Survey participates in evaluating fundamental rock properties, rock structures, lithostatic pressures, stability, and other related issues. The Survey is conducting basic experimental studies to determine the likely interactions among salt, brine, canisters, and waste over a range of temperatures and pressures that may be considered in disposal of high-level radioactive waste or spent fuel. In addition, the Survey is cooperating in a brine migration experiment being conducted at Avery Island in Louisiana. Individual USGS scientists also participate in peer review committees such as the BWIP Geology and Hydrology Overview Committees.

III.D.2.1.2 Bureau of Land Management

The Bureau of Land Management (BLM) of the Department of the Interior has the responsibility of overseeing and controlling the use of certain Federal lands. Where exploration activities are conducted on such land, the BLM is contacted and permission secured. As necessary, a formal cooperative agreement will be prepared jointly by BLM and the Department to document what activities are to be performed, how those activities are to be conducted, and conditions for land restoration.

Interactions between the Department and the Bureau already have taken place concerning site characterization activities in the Paradox Basin in Utah and at the Los Medanos Site in New Mexico (37).

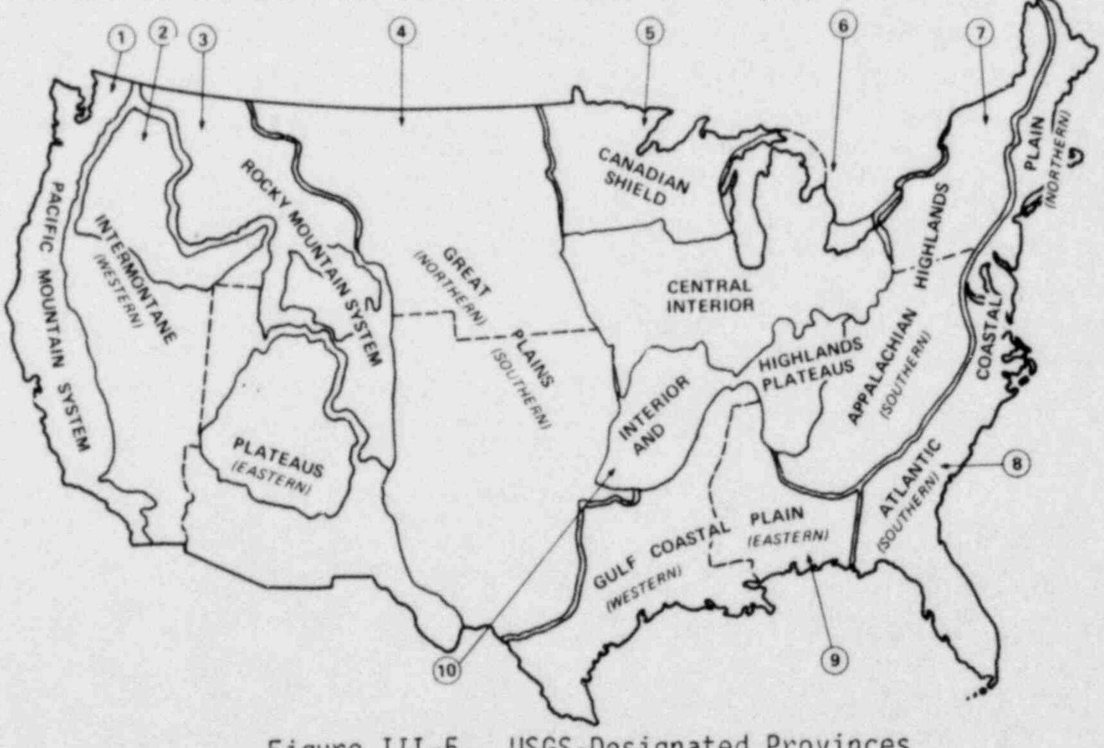


Figure III-5. USGS-Designated Provinces

Source: (Reference 36) Adapted from N.M. Fenneman, "Physiographic Divisions of the United States," Annals of the Association of American Geographers, 3rd edition, Vol. 18, pp. 261-353, 1928.

III.D.2.2 Corps of Engineers

The Corps of Engineers has the most experience within the Federal Government regarding real property acquisition. Therefore, limited working relationships with the Corps of Engineers have been established (38) to assist the NWTS Program. Future expansion of Corps of Engineers involvement to support siting of a repository or more extensive R&D activities could readily be achieved. Their services have been and are being used to obtain access for field activities in Louisiana. Its current duties in Louisiana include:

1. Determination of land ownership and holders of surface and subsurface rights from whom permission must be secured in order to enter property for the purpose of field exploration and, in the future, for acquisition of a repository site.

2. Contacting the landowners and negotiating rights-of-entry, leases, or other legal instruments as required for land access.

3. Making payments to landowners for leases that have been obtained.

The Corps of Engineers Waterways Experiment Station in Vicksburg, Mississippi, is investigating the composition, constitution, properties, and interactions of materials considered for use in plugging boreholes and sealing shafts (39).

III.D.2.3 U.S. Department of Agriculture

The U.S. Department of Agriculture (USDA) has responsibility for access to National Forest lands. To implement the Department's program it has been necessary to implement agreements to allow investigations to continue while providing full protection to the environment.

A Memorandum of Understanding (40) between the USDA and the Department was prepared for geological/hydrological characterization activities in the De Soto National Forest of Mississippi for the Gulf Coast Salt Dome Study.

The USDA Science and Education Administration, through a series of Land Grant Colleges, can provide a socioeconomic impacts determination and mitigation program. This program provides site-specific methodologies to assess socioeconomic impacts and to analyze potential mitigative actions. USDA is also establishing a Technical Advisory Panel for periodic peer review and evaluation of these studies. An interagency agreement has been negotiated to bring this USDA expertise to the program will shortly be signed.

III.D.2.4 Summary

Establishment of nuclear waste management policy by the President has established a strong foundation for interaction among the appropriate Federal agencies. The existing cooperation can be expected to expand under the President's strong interest. As this coordination continues, technical, administrative, and regulatory delays will tend to be minimized to reduce impacts on the schedule of the overall NWTS Program.

III.D.3 Land Acquisition Activities

Where the Department does not already own or control a proposed repository site, the acquisition of the real property for the repository must be included in the schedule. Non-Federal land can be acquired by the Department for a repository site (41), following procedures already established within the Department (42). Federal property controlled by other agencies may be acquired by transfer to the Department following procedures established by the General Services Administration (43) or by the Department of the Interior.

Public lands administered by the Secretary of the Interior through the Bureau of Land Management can be used for a repository only after withdrawal of such lands (44).

The Secretary of the Interior may, upon request by the Department, administratively withdraw public lands of 5,000 acres or more for a period of up to 20 years (45). The Department of the Interior has proposed regulations that would permit renewals of such a withdrawal (46). However, the administrative procedure would not be practical for the withdrawal of lands for a repository, which entails permanent and exclusive use of these lands.

If a site of 5,000 acres or more is proposed on public lands, the Department plans to request administrative withdrawal of such lands for a term not to exceed 20 years. During the time of the administrative withdrawal, required site characterization and other preselection activities will take place. If the site is ultimately selected for a repository, the Department will seek to obtain permanent withdrawal of the site by statute.

The Department proposes to acquire an interest in a site for a repository following the designation of a preferred site in a region. This action, as part of the site banking process, will occur in parallel with continuing site characterization activities. In this way, and as specifically demonstrated by the above outline of the Department's plan for the withdrawal of public lands, any potential impact of property acquisition on the schedule would be minimized.

III.D.4 Availability of Expert Staff

III.D.4.1 Introduction

Skilled technical personnel will be required in the site exploration and characterization phase of the program, in the development of necessary technology, and in the design and construction phase of the repository.

III.D.4.2 Site Exploration and Characterization

The site exploration and characterization phase of the NWTS Program requires the use of expert human resources in various disciplines, particularly in the area of earth sciences. The availability of these human resources has been evidenced by responses received from solicitations for organizations to perform geologic project management and site characterization and exploration work. For example, 24 firms responded to the original (1977) solicitation for the selection of geologic project managers for regions currently being investigated. A recent solicitation for an additional geologic project manager elicited 12 responses from qualified organizations. Other efforts in FY 1980 involved solicitations for a national site screening project and elicited 6 responses. For deep drilling in the Permian Basin, 6 responses are expected; and for an environmental project manager, 10 responses are expected. The responses to a variety of site characterization and exploration solicitations indicate an adequate resource capability for conducting these activities in a timely manner, even though other energy-related activities may concurrently be receiving attention.

III.D.4.3 Repository Design and Construction

Through prior repository conceptual design activities (34, 47-49), several of the nation's largest engineering firms are familiar with the requirements for repository design and construction. Typically, the engineering and scientific disciplines involved in the design of a repository include those shown in Table III-1. Improved capabilities in both the nuclear and mining engineering fields over the past decade mean that many engineering firms today are capable of designing a repository. Previous procurement activities for conceptual design have resulted in expressions of interest from over 20 major engineering firms, most of which have been considered qualified to perform the work. Design support contractors in specialty areas, e.g., rock mechanics, radiation shielding, and environmental effects, are becoming increasingly available. The earth sciences technical community is now and will continue to be heavily involved in the conduct of the program.

From a construction standpoint, virtually all the expertise required for a repository is currently available. Construction skills may be in limited supply in the field of shaft sinking, where a rapidly expanding minerals exploration industry causes high demand for such services. However, this limitation can be overcome by advance procurement planning. Construction requirements will be known sufficiently in advance of the time of construction, chiefly because of the length of time required for licensing reviews prior to the issuance of the Construction Authorization. Between the time an application is made to the Nuclear Regulatory Commission and the time construction authorization is issued, the Department will be able to engage all necessary construction skills.

III.D.4.4 Operating Staff

Present planning calls for engaging the repository's operating contractor prior to or coincident with submission of the license application, to ensure that operational considerations are properly factored into the design and construction. This strategy allows the operating contractor to be available and allows the development of operating staff for many years in advance of actually beginning receipt of waste shipments. Thus ample opportunity is available for developing the necessary expertise of the operating staff. Opportunities for operator training exist at the repository site itself, other Department installations, and through their participation in functional checking and acceptance testing of repository systems.

III.D.4.5 Summary

The design and construction expertise required to build a mined geologic repository is currently available in the United States. Operating expertise will be available by the time the repository is ready for waste emplacement. The schedule for implementing the repository option should not be affected by the need for lead time to develop the needed expertise.

III.D.5 Logistical and Administrative Factors

Although sufficient design and construction expertise exists, plans to engage specific contractors for the first repository have not been completed. Major contractors involved will include the architect-engineer, the construction manager, and the repository operating contractor.

Table III-1. Specialty Disciplines Required for Repository Design

Underground	Surface	General
Shaft sinking	Remote handling equipment and techniques	Thermal analyses
Hoisting systems	Hot cell design	Shielding analyses
Mine design and planning	Remote viewing systems	System studies
Mine construction	Decontamination technology	Licensing experience
Mine ventilation	Waste processing (in-plant)	Civil engineering
Bulk materials handling	Waste processing (interfaces)	Safety analysis
Operations analysis/logistics	Ventilation for containment	Quality assurance
Shielding analyses	Complex mechanisms	Reliability assurance
Thermal analyses	Assay and NDT technology	Scheduling
Rock mechanics	Operations analysis/logistics	Estimating
Hydrology	Seismic and tornado analysis	Architecture
Geology	Shielding analyses	Process piping
Geochemistry	Thermal analyses	Electrical engineering
	Bulk materials handling systems	Instrumentation
	Railroad construction	Controls
	Mitigation of environmental impact	Safeguards/security
		Environmental engineering
		Nuclear engineering

As all of these contract tasks involve large amounts of money, with activities spanning several years, Federal procurement regulations requiring competitive bidding apply. The procurement process may take as long as 12-15 months on each contract, as shown in Table III-2. Current planning for repository implementation recognizes these long procurement intervals, and consequently these intervals are not viewed as critical to the overall schedule. Consideration will be given to engaging the architect-engineer contractor at an early date, as candidate sites are qualified. Architect-engineer contractors are currently engaged to conduct special studies and evaluate engineering options for repositories in salt and to develop a conceptual design for a repository in basalt. Construction managers and construction contractors can be sought and acquired while the Commission is reviewing the license applications. The operating contractor normally should be selected prior to start of construction; however, such a contractor could be sought as soon as the repository site is selected and then could provide input to the design process and to the licensing process.

The repository is defined as a "Major System Acquisition" within the Department. As such, formalized procedures govern its development. These procedures specify the need for developing system missions and charters, along with project management plans, at an early date. An Energy Systems Acquisition Project Plan (50), which outlines administrative logistical factors as discussed above, has been prepared and approved within the Department.

The design and construction of the repository will be accomplished with funding specifically authorized by Congress as part of capital budgets. Requests for capital funding are submitted by the Department when and as required. Under current planning assumptions, [illegible]ority may be necessary by FY 1983 to ensure banking or protection of sit[illegible]. Timely execution of the repository project will depend on necessary authorizations of funds by the Congress. The potential impact on the schedule of deferred or fragmentary funding authorization is impossible to assess in advance. However, as part of the planning of construction projects, scheduling and estimating techniques are used at the preauthorization stage to allow rapid assessments of schedule slippages due to budget reductions. Careful planning of funding requests and procurements will minimize schedule slippages resulting from funding delays.

Tab. .I-2. Typical Department Procurement Process for Cost-Type Contracts in Excess of $10,000,000

Action	Elapsed Time (calendar days)
Establishment of Contractor Evaluation Board by Source Selection Official	0
Preparation of Selection Criteria	7
Approval of announcement to be published in Commerce Business Daily and Selection Criteria	30
Publication of CBD announcement	37
Receipt of Standard Forms 254 and 255	68
Board evaluation of Standard Forms 254 and 255	79
Formal request for supplemental information sent to qualified firms	110
Discussions with top-ranked firms	141
Board recommendations/presentation made to Source Selection Official	171
Firm Selection	185
Cost proposal requested from contractor	215
Audit completed	265
Negotiations completed	287
Contract to Department Headquarters for approval	317
Contract signed	347

III.D.6 Design and Construction Time Factors

III.D.6.1 Introduction

As a result of conceptual design activities conducted to date (1, 2), the details of design and construction scheduling have been developed for a typical repository. These schedules are used as a reference case in this discussion, and assessments of schedule differentials associated with building repositories in various geologic formations are made.

In developing schedules for repository construction, it is significant that, although a repository has not yet been built, numerous examples exist for the construction of facilities and projects that are quite similar to elements of a repository. Experience exists from the construction of nuclear receiving facilities, and from shafts and mine development throughout the United States. Consequently, there is a basis to support the planning assumptions. It is also of note that surface facilities at the repository for handling radioactive materials are, by comparison to a reactor or a reprocessing plant, relatively straightforward.

In the following discussion, a typical design and construction schedule for a salt repository (47, 48), ramifications of other media (49), and potentials for delay that may exist, are addressed.

III.D.6.2 A Typical Salt Repository Schedule

Figure III-6 presents a representative engineering schedule and construction schedule from a recent study (47) done for a dome salt repository.

Initial engineering activities would, over a 2-year period, develop sufficient design knowledge to support the license application. Current planning allows for this activity to initially address multiple specific sites; final site selection would be decided during the first year of engineering work.

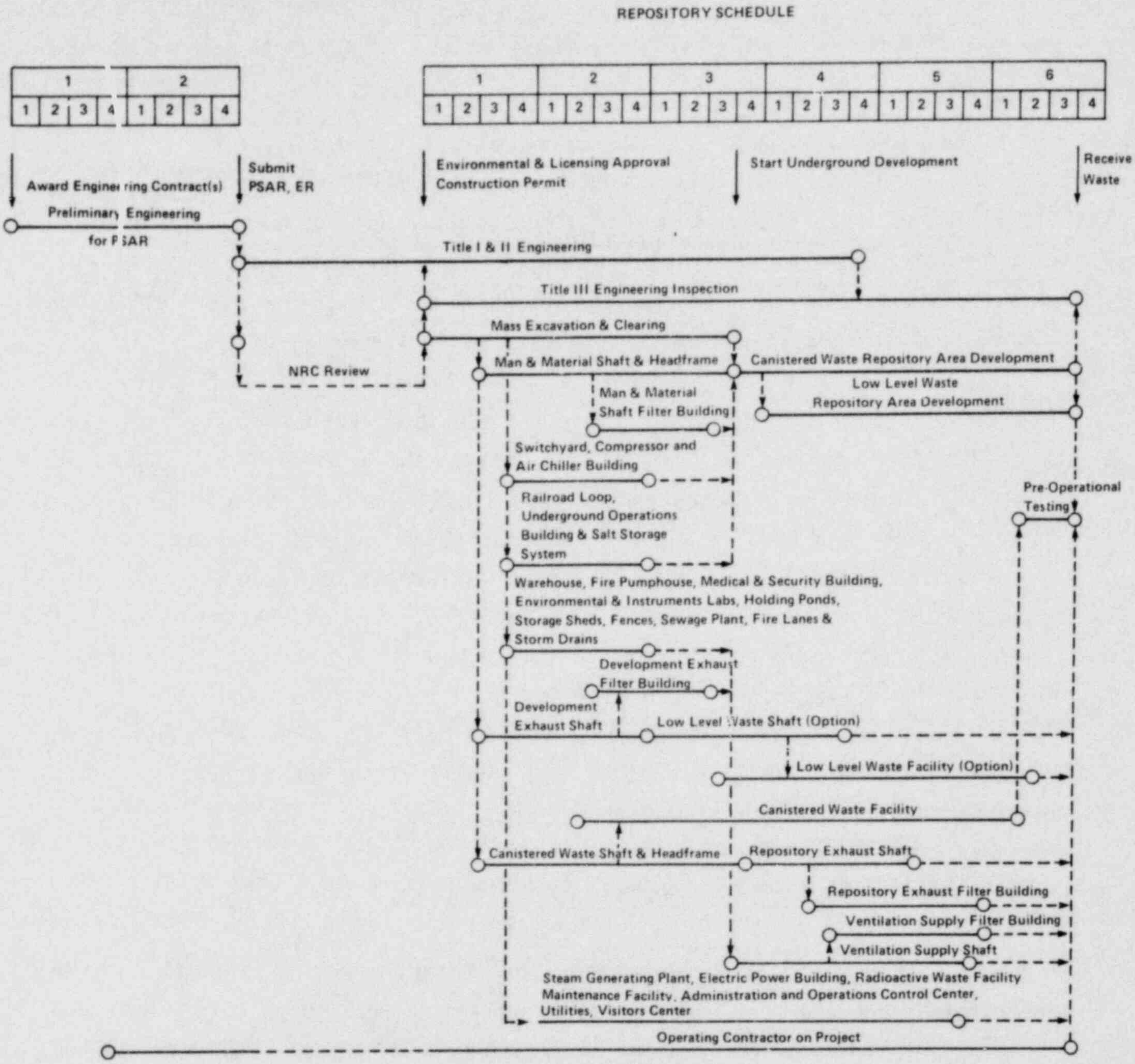


Figure III-6. Standard Construction Schedule

Source: (Reference 47) Stearns-Roger Engineering Company, Conceptual Design Report, National Waste Terminal Storage Repository for Storing Reprocessing Wastes in a Dome Salt Formation, DOE Contract No. EY-77-C-05-5367, U.S. Department of Energy, January 1979

After making the license application, Title I design for surface and subsurface facilities will start, supported by project-specific capital funding. Title II Detailed Design (also supported by capital funding), follows Title I. The shift from Title I to Title II will occur at different times for each element of the repository. Design efforts will initially concentrate on those elements of the repository that either will be constructed first or have long procurement intervals.

A minimum of 2 years of Title I and II design will be required before construction activities can begin. This time is required for developing sufficient Title II design for critical elements of construction such as site development and shafts, so that construction bids can be solicited and contracts awarded for those elements of the repository. This design work can proceed in parallel with the licensing review process. Completion of Title II design for the balance of repository facilities can proceed concurrently with construction of previously designed critical components and can be sequenced to the overall construction plans as required.

Construction and checkout of the facility will require about 6 to 10 years (69 to 117 months) depending on the rock type, at which point the repository will be ready to begin receiving wastes. Construction of the shafts and development of the waste emplacement building above the main shaft and development of the underground area are the critical activities that will determine the time required to build the repository. The construction of surface facilities can be sequenced to support underground development, minimize congestion and interference among adjacent facilities during construction, provide for an orderly transition from construction to operation, and maintain reasonable manpower levels.

Assumptions have been made in these schedules regarding such factors as climate, transportation, labor availability, and productivity for repository design that are consistent with common construction practice.

Approximately 8 years will be required to design and construct a repository in salt. Many separate organizations, government agencies, architect-engineers, and contractors, will participate in the design and construction of the project. Effective management will be a determining factor in successfully meeting such a schedule. The major features of the schedule are summarized below and discussed in subsequent paragraphs.

Titles I and II design required prior to starting construction (concurrent with licensing review)		24 months
Initial site clearing and excavation	6 months	
Initial shaft sinking and support facilities	27 months	
Underground development, completion of shafts and surface facilities	36 months	
Construction duration		69 months
Total duration		93 months

Engineering prior to construction will include those activities necessary to let the initial construction contracts and develop standards for initial subproject engineering packages.

An estimated 6 months will be required to prepare the site for initial construction activities. During this period, temporary site access routes will be developed, temporary utilities acquired, and the area surrounding the shafts cleared and brought to final grade. The scope of work and time required for initial site development will be determined by site conditions and could change significantly when a specific site is selected.

Three shafts will be sunk first, followed immediately by the other remaining shafts. Shafts for men-and-materials access and the mine development exhaust shaft are required first for underground development. Early completion of the canistered waste shaft is desirable, so that construction of the canistered waste handling facility can be started as early as possible. Conditions at a specific site could affect the time required to sink shafts. At this phase, ancillary surface facilities to support mine development and construction of major surface structures will also be constructed.

Sinking the men-and-materials shafts, constructing the waste receiving building, and developing the canistered waste emplacement area are the critical activities in this schedule. They essentially determine the duration of the schedule and establish the general time frames available for completion of other activities. The waste emplacement area developed during construction could be sufficiently large to accommodate 5 years of waste receipts (approximately 9,000 metric tons). This is approximately 15%-20% of the total area that will eventually be developed underground.

Surface facilities are scheduled to start as soon as it is possible to do so without interfering with construction of more critical facilities. This creates a schedule which allows most facilities to start anywhere within a given duration frame rather than on a specific date. This flexibility will enable schedules to be structured to meet limitations in resources or allow for special circumstances. Construction of facilities will be scheduled to produce a reasonable manpower distribution.

III.D.6.3 Implications of Media Other Than Salt

Salt has the property of being relatively easy to mine in comparision with other rock types. Elaborate roof support systems are generally not required, and mechanized mining systems can be employed to allow fairly rapid excavation.

This is not the case for many other geologic media under consideration. Repositories in argillaceous rocks may require extensive roof and wall support systems (51). Basalts and granite rocks may require the use of different mining methods, so that the time required for excavation of a given area may increase. The amount of excavation needed to accommodate a given quantity of waste is dependent on allowable thermal loading criteria. For the specific case of basalt, the incremental difficulty in excavation (with respect to salt) is balanced by higher allowable thermal loadings. Thus current estimates for the construction time for a basalt repository of an equivalent capacity would be comparable to the estimate for repositories in salt.

Salt and basalt represent two geologic media most studied to date in the program. Extrapolation of construction requirements to other geologic media is difficult without detailed knowledge of the media. Nonetheless, using the results of preliminary studies (49), it is concluded that a repository can be constructed and checked out in alternate media within 8 years. This estimate is reasonable for planning purposes, and is longer than expected schedules of 69 months for salt and basalt formations.

III.D.6.4 Potential Delaying Factors

Prudence requires that, as a project goes forward, attention be given to verifying that the planning assumptions and their rationale are not altered. Situations may be encountered that will cause reassessment of construction schedules. Therefore, several sources of potential delay have been identified and their potential impact discussed below.

III.D.6.4.1 Procurement Delays

Current design programs have identified major components of the repository with long procurement lead times. These are factored into current schedules, and therefore do not represent a major perturbation. The basis for this position is that long-lead items get early attention during the design process. Currently it is estimated that 2 years of design time are necessary prior to construction, and during this time emphasis is given to long-lead items. This time interval is provided for in the schedule.

III.D.6.4.2 Strikes and Labor Disputes

Strikes and labor disputes can, if not considered in the planning phases of the construction, disrupt construction schedules. Construction contractors for the repository will be encouraged to utilize collective bargaining techniques that minimize the potential for such actions. Though no specific allowance has been identified in the schedule, an allowance for lost time comparable to that experienced for other projects of this size is included.

III.D.6.4.3 Current Schedules

Current schedules call for simultaneous commencement of construction for several shafts and other elements of the repository. If instead, the Department elects to take an approach wherein the full construction effort starts only upon completion of the first shaft, the schedule would be extended about 3 years. Figure III-7 depicts a construction sequence for this approach. While the present strategy does not envision this action, it is acknowledged as an alternative.

III.D.6.4.4 Implications of an Exploratory Shaft

If construction of a shaft for exploration at depth were required before submission of a license application, the start of design and construction would be delayed about 40 months. However, the duration of full-scale repository construction would not be significantly affected as multiple shaft excavations are presently planned.

III.D.6.5 Summary

Of the many factors that can influence construction schedules, the majority are associated with specific site conditions and environment, and are, therefore, resolvable as the site selection processes evolve. Other factors are associated with strategic changes to repository planning, and thus represent a variant that comes from both site conditions and regulatory consideration.

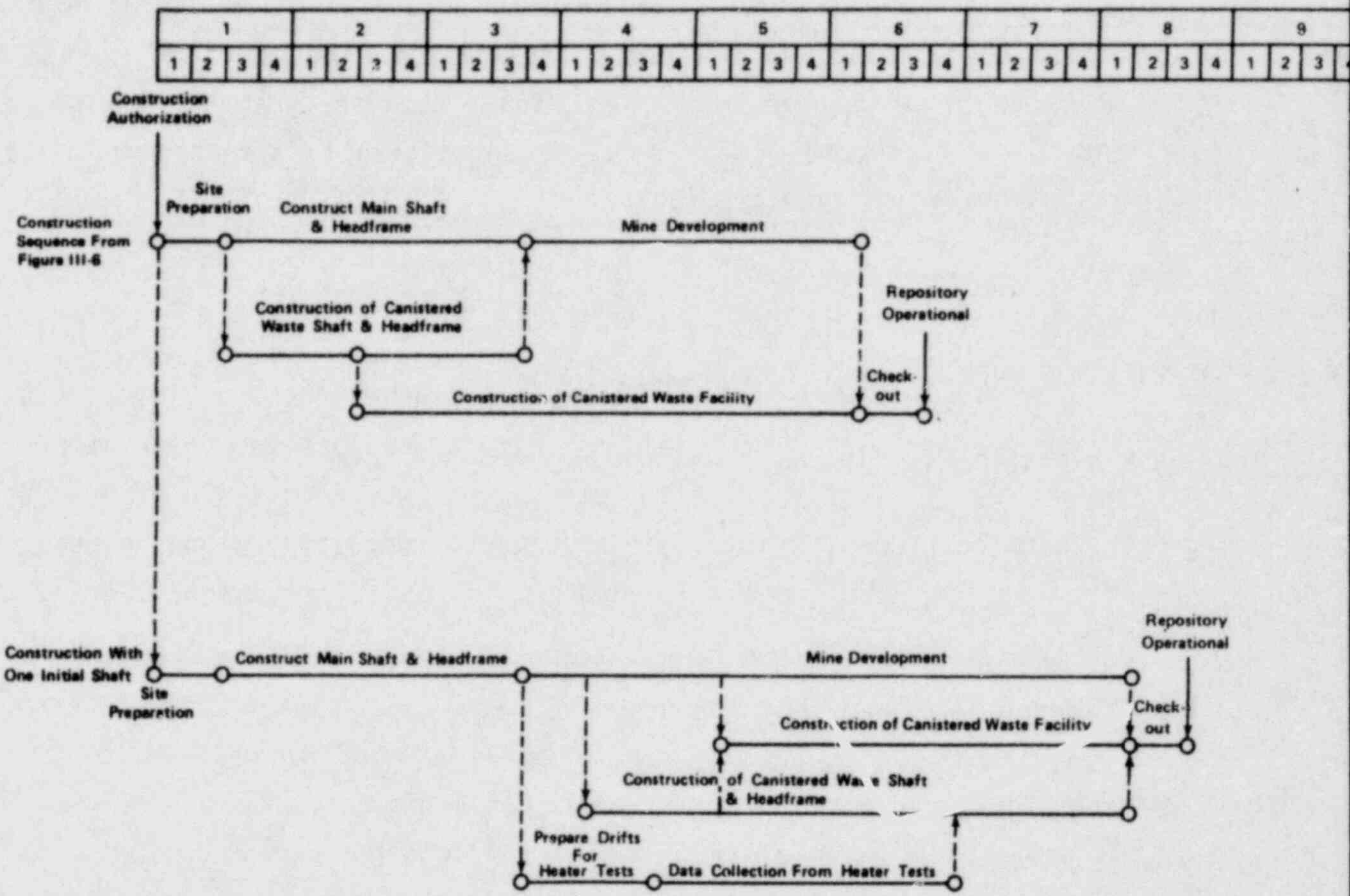


Figure III-7. Extended Construction Schedule

Source: (Reference 47) Stearns-Roger Engineering Company, Conceptual Design Report, National Waste Terminal Storage Repository for Storing Reprocessing Wastes in a Dome Salt Formation, DOE Contract No. EY-77-C-05-5367, U.S. Department of Energy, January, 1979

III.D.7 Initial Operation Period and Backfill

The objectives of the initial repository operation period are to (i) adequately demonstrate the full-scale handling and emplacement of wastes, particularly those with high thermal emissions, in an underground environment; and (ii) confirm underground repository design and performance assessments.

The successful handling of a few thousand canisters will demonstrate the capability of all aspects of the waste handling and placement systems.

The problems of confirming underground designs and performance are more involved. The goal is to verify the predictive capability of the methods used to apply early geologic test data to the specific site and design configuration and to ensure that no phenomena associated with actual waste placement have been overlooked. To accomplish this goal, extensive instrumentation and testing are planned for the first few years of operation.

Based on the types of activities involved in the initial operation, it is possible to derive and support certain conclusions with respect to repository design confirmation. First, the experimental period is not intended to verify rigorously the performance of the repository over the very long term. Instead, its function is to provide in situ data, at the specific site, on real waste, over a reasonably large area, as additional input to be issued in mechanistic computational models which will have been used to make the long-term projections upon which repository licensing will be based. Additional observations made during the initial operation will be used to confirm previous long-term predictions. The other principal function of the initial period is to provide an opportunity to identify any processes or phenomena that may have been overlooked in the initial analysis. Current planning indicates that approximately 5 years will be necessary for this period. Accelerated tests, such as those discussed in II.E.1, can be performed to effectively duplicate repository conditions that might develop over long periods of time, so that decisions to modify design features or procedures can be made after a few years of operation.

These early years of operation are expected to feature methods and mechanisms that allow ready retrieval of the waste. Examples of such mechanisms include maintaining the storage rooms in an open state and inserting protective sleeves around waste canisters to mitigate interactions with the host rock.

In the event that repository operation or long-term isolation is seriously affected by conditions previously unforeseen or analyzed, an alternate repository would need to be considered. Current planning involving multiple repositories schedules these repositories to become available about every 3 years following the initial facility. Thus a backup capability is expected to be readily available within the initial operational period. The need for retrieval of the repository's inventory will be based on an overall safety comparison of the risk of retrieval versus the risk associated with leaving the waste in place.

Upon completion of the initial experimental period, reliance on easy retrieval by the abovementioned means could be abandoned with the approval of the NRC. Canister-filled rooms could then be backfilled as soon as practical after the experimental period. Studies (34, 51-53) have demonstrated that retrieval of the waste from backfilled rooms is technically possible. It is believed that such a need will not develop, however, because by then the entire formation will have been extensively mapped and tested. Nonetheless, as long as access from surface to the underground level is maintained, exhumation of large fractions of the inventory will be possible. As with retrieval in the early phase, exercising a retrieval option at any time prior to closure will have been predetermined in consultation with the Commission as part of its review and approval of the decommissioning plan.

It is unlikely that including provisions for retrievability will affect the schedule of repository development. Ample latitude is provided for methodical, step-wise development including testing and evaluation. A high level of confidence concerning the integrity of the operation will be attained before backfilling will commence. Should retrieval of waste be necessary following the initiation of backfilling, waste management plans include rerouting the wastes to other facilities.

III.D.8 Technology Development

III.D.8.1 Introduction

Technical developments are needed and planned to ensure that the required technological basis is available to meet the schedule for repository development. The current status of NWTS programs has been discussed previously in II.D.4, II.E.1 and .2, and II.F.1 and .2. This section briefly describes pertinent technology development activities for the waste package (54), repository, and site subprograms (55) and gives the anticipated schedule for these activities. The Department is proceeding with a systematic program to develop the needed technology on a timely basis. The relationship of major technology development milestones to the overall program schedule maybe seen in Figures III-2 and III-3.

III.D.8.2 Waste Package

The Waste Package Subprogram is directed toward development of waste packages acceptable for use in a mined repository in various media; its status is described in II.E.1. The development program ranges from an analysis of the rationale for the waste package function to verification that it indeed performs that function. Criteria for the waste package function are being established for all package components, including the candidate waste forms; materials are being evaluated and selected; and package designs are being developed. The technical approach emphasizes the following areas:

1. Waste forms are being developed and evaluated with emphasis on spent fuel. Models to predict long-term spent fuel behavior will enable evaluation of alternative waste forms as well. Near-field models, including package assessment capability, will be operational in 1982, and assessments with integrated models will be possible in 1983.

2. Barrier materials which form the components of the waste package are being developed and evaluated. This activity covers materials for the canister, overpack, and hole liner or sleeve, as well as stabilizers and emplacement hole backfill materials. Material screening for package materials appropriate for salt will be completed in 1980. Screening for additional materials which might be more appropriate to other host rocks is continuing.

3. Waste package development includes design, analysis, testing, and qualification of package components and complete engineering scale packages. Laboratory and field studies in selected geologic test facilities of reactions among waste package and repository components will be carried out and the results iterated with design activities. These activities will lead to waste package conceptual designs for salt and hard rock in 1981 and 1982, respectively. Appropriate package design concepts will be selected for salt and hard rock in 1982 and 1983 respectively. The first site-specific design will be complete in 1984. Other site-specific designs will follow as sites are selected.

4. Verification tests of complete fabricated packages are planned to assess handling techniques and short term performance of the qualified waste package. These tests will be conducted underground in field test facilities, and ultimately, monitored during start-up at specific repository site locations. Preliminary field data for verifying waste rock interaction models and waste package components as well as a recommended suite of in situ confirmation tests at the selected repository site will be available in 1987.

III.D.8.3 Repository

The Repository Subprogram is directed toward the development of a viable underground civil structures and above ground facilities for the isolation of nuclear wastes and for the necessary equipment and instruments to operate and decommission the repository, as discussed in II.E.2.. Components of the repository serve as additional barriers between the waste package and the surrounding rock strata. The technology development subprogram to support repository engineering emphasizes the following activities:

1. Analyses will be carried out to establish reference repository conditions for granite, salt, and basalt. Key elements to be defined in the analyses include thermal, mechanical, hydrological, and chemical parameters. The analysis will be complete in 1981.

2. A data base will be developed through experiments (laboratory and field) to provide fundamental information in support of repository design and performance modeling. This data base will be derived from studies in rock mechanics and waste-rock interactions. Such studies are a continuing activity and are well under way. Data sufficient for initial design will be available in 1985.

3. Instrumentation will be developed to provide monitoring of repository status during testing and operational phases. Such instrumentation will be developed and available for in situ testing in 1987.

4. A test program will be conducted to provide the basis for selection of materials and design of plugs and seals for shafts and boreholes, as well as chamber backfills. Plugging material studies are being conducted on laboratory and field scales to establish seal lifetime in various rock types. Designs for repository seals will be available for bedded salt in 1983, domed salt in 1984, and granite in 1985. Final design for the chosen site will be available in 1987.

5. Information will be developed to enable evaluation of consequences of natural phenomena, e.g., earthquakes, on repository design and will be completed in 1981.

III.D.8.4 Site

Selection of a proper site for the repository is critical to the whole concept of a mined repository and is discussed in II.D and III.C. Technology support to site selection is twofold: first, a geohydrologic scientific and technological data base is being developed, and second, a capability for overall repository/site performance assessment is being developed and will be verified and applied. The technology development subprogram to support site selection emphasizes the following activities:

1. The development of a data base on media properties that influence radioactivity migration/retardation is under way in the laboratory. Nuclide migration tests will be conducted with radioactive material in tuff and granite at the Nevada Test Site. A main objective of these tests is the development of test methods for evaluating nuclide transport in the field. Results from those tests will be available in 1983 along with data for comparison to laboratory sorption coefficients and for model verification. Data sufficient for site selection will be available in 1985.

2. Characterization of geologic samples from candidate sites with respect to thermal, chemical, and mechanical properties will be available for site selection in 1985.

3. Development of integrated models for the performance assessments of the repository system with respect to radioactivity release scenarios and their consequences will be available for use in 1983. Models for evaluation, if necessary, of thermal-mechanical-hydrologic coupling will be available in 1984.

4. Verification of models will be based on the results of specifically designed laboratory experiments and field tests and by model application to simulate certain natural phenomena. Underground testing of waste/host rock interactions are being started this year, additional nuclide transport studies in field test facilities will be initiated for waste package components, and a recommended suite of in situ site confirmation tests will be available in 1987.

5. Application of the performance model to site evaluation will start in 1981 and its use will continue throughout the site qualification and licensing process.

III.E COSTS OF MINED GEOLOGIC DISPOSAL FACILITIES

III.E.1 Introduction

This chapter addresses current estimates of the costs of geologic disposal of spent fuel. Costs for research and development, repository facilities, and packaging/encapsulation facilities are identified. The existing basis for developing repository component costs is described, and relevant studies are summarized. Representative repository component costs are displayed, and the method used to develop these cost figures is presented. Subsequent discussion then illustrates the sensitivity of repository costs to variations in key design parameters. Repository costs presented herein are then compared to previously published costs, and cost differences are explained.

Cost considerations associated with options for interim storage of spent fuel and other elements of a total waste management system are not specifically addressed here. A discussion of storage and other related costs appears in Part V, and summary discussions in Part VI address waste management costs in the context of the cost of power generation.

III.E.2 Research and Development Costs

Estimated R&D costs for encapsulation and mined geologic disposal of spent fuel from commercial power reactors recently have been developed (56). The total R&D cost will approach $2 billion. This estimate includes studies in salt, basalt, volcanic tuff, and granite; waste package development; continued study of an alternative disposal concept (subseabed disposal); and equipment expenses.

III.E.3 Mined Geologic Repository Costs

During the past 2 years, seven significant studies (34,47, 48, 57-59) addressing repository economics have been performed. The first two studies (47, 48) are detailed repository conceptual design studies independently done by major engineering firms. Their results also are reported in a special study to reconcile differences in their cost estimates (57). These two conceptual design repository estimates are for different types of salt media (domed or bedded) and for different waste forms placed in each (high-level waste and spent fuel).

The spent-fuel disposal study (58) draws heavily on the conceptual designs (47, 48). Principal variations in repository economics are based on differences in waste form (reprocessing waste, encapsulated spent fuel, chopped spent fuel, and vitrified spent fuel). Cost categories are similar to those used in the EIS. Several variations of each waste form are studied with respect to other technical parameters without preparing costs for each alternative.

The draft environmental impact statement on commercial waste management (34) also provides an original independent cost estimate. However, this EIS is based on a less detailed cost breakdown and a much wider estimated range of scenarios: for example, 16 combinations of waste form and emplacement media were studied for commercial waste repositories.

The Office of Waste Isolation Technical Manual-36 (59) contains another original independent detailed repository cost estimate. Variations include waste form, waste container, retrievability options, and media. In all, cost estimates are prepared for 18 cases.

The repository included in the Department's preliminary spent fuel acceptance charge estimate (60) is based on a planning study undertaken prior to the completion of the bedded salt conceptual design cost estimate (48). Only the base line repository cost is reported.

III.E.3.1 Repository Component Costs

This subsection defines components of repository costs for design and construction, operations, and decommissioning. Conservative estimates are developed for repositories in various geologic media. These repositories are "standardized" in terms of using identical facilities and criteria wherever possible. The sensitivity of repository costs is discussed for 3 variations: rock type, thermal loading limits, and repository size.

III.E.3.1.1 Basic Assumptions

For cost estimate purposes, it is assumed that the repositories are based on the conceptual designs (47, 48) and preliminary judgments for most likely engineering concepts. All repositories include facilities to receive, inspect, emplace, and retrieve spent fuel and perform the associated underground excavation and support activities. Following their operational life, the repositories will be decommissioned, including mine backfilling, shaft sealing, surface facility demolition, and restoration of surface site.

The repositories will initially be operated in a readily retrievable mode, during which time the emplaced fuel can be easily removed by planned operational procedures. The receiving rate is limited to 1,800 MTU/yr during the first 5 years. Following this period, the receiving rate capability is increased to 6,000 MTU/yr. During this period, methods used to enhance ready retrievability are no longer utilized. Total storage capacity of a particular repository depends on the usable area of the geologic formation and the thermal loading criteria, both dependent on site specific conditions.

The basic surface facilities will not vary significantly because of type of geologic media. However, mining and emplacement details can change for different media and site conditions. For example, owing to their overall cross-sections, small salt domes would require special considerations in mine development planning. The drilling of shafts will vary according to stratigraphy and the depth at which emplacement occurs. Mining methods and emplacement position preparations will differ with the difficulty of excavation of the rock. These effects are taken into account through use of factors originally developed as backup data.

The estimates presented here were prepared as composites of existing designs and estimates, comparing similar facilities among various designs and selecting a conservative concept as a standard for use in all media. Media-specific costs, such as shaft sinking and excavation, were derived for basalt and granite from detailed salt estimates by factoring to consider rock density and ease of excavation, using factors previously developed (59). Mine development configurations are assumed identical for all media.

Where a standard component was selected from a particular design, the component's attendant operating and decommissioning costs were also used. Where this was not the case, factoring as outlined above, was utilized.

Site-specific influences on cost (e.g., remoteness from utilities and roads) were assessed on a best-case/worst-case basis, using the methods in the Cost Estimate Reconciliation Study (57). The estimates presented consider the worst case for each of these constraints.

III.E.3.1.2 Design and Construction

Costs incurred under this category include all preliminary and detailed engineering design performed on a site-specific basis; construction of all surface facilities, shafts, and subsurface support areas; excavation of a limited number of storage rooms (enough to accommodate 5 years worth of receipts); engineering inspection of all construction; and functional acceptance testing and run-in of completed systems.

III.E.3.1.3 Replacement of Capital Equipment

This category includes expenses for replacement of major equipment items such as mining machinery, hoist motors, and waste transporters that will be necessary during the operating phase of the repository.

III.E.3.1.4 Operations

These costs include operating costs of facilities for the receipt, handling, and storage of canistered spent fuel assemblies from both BWR and PWR commercial power plants. The estimate includes all operating costs for a repository, as well as mine development beyond the first 5 years and costs for operating contractor key personnel for planning, testing, training, and maintenance during the line item-construction period.

Both fixed and variable categories of operating cost are included. Fixed operating cost is associated with keeping the repository functional even if the receiving rate is zero (e.g., a supervisory work force). These costs typically accrue as annual costs. Variable operating cost is dependent upon receiving rate and includes labor and materials associated with receiving, processing, and emplacement (including excavation for emplacement). Operating costs end with the completion of storage room backfilling.

III.E.3.1.5 Decommissioning

This estimate includes all decommissioning costs for a repository constructed in accordance with the contemplated design and decommissioned in accordance with the program outlined in the referenced reports. Costs for operating contractor personnel on the site during decommissioning are included in the referenced report and not in the operating costs. The three major elements of decommissioning were assumed to be demolition of surface facilities, backfilling of main entries and airways, and shaft sealing.

III.E.3.2 Factors Affecting Repository Costs

The assumptions used for key repository design parameters vary widely in the studies (34, 47, 48, 57-60). Choice of emplacement media, thermal loading limits, and repository underground size all influence repository costs significantly. The following discusses each of these in turn to highlight their effects on repository costs.

Tables III-3 through III-5 present representative repository costs as functions of emplacement media, thermal loadings, and repository underground size. These costs do not include spent fuel packaging facility costs, which are shown in Table III-6. Values for these parameters are identified along with capacities and operational periods associated with the estimates. To aid the discussion, undiscounted unit costs are presented, representing the quotient of total cost divided by total receipts, expressed in dollars per kilogram of heavy metal. This figure, although useful to show comparative costs, is not the same as the component which would be included in a charge for storage and disposal of spent fuel, as no discounting or estimates of future inflation are included.

III.E.3.2.1 Alternative Geologic Media

The choice of geologic medium influences costs. Although salt has been the principal focus for most past cost studies, two studies (34, 59) also assess basalt, granite, and shale repositories. Table III-3 presents illustrative cost figures for repositories in dome salt, bedded salt, basalt, and granite. These figures assume that repository thermal loading limits are constant at 40 kW/acre. In actual practice, determination of thermal loading limits will be based on site-specific conditions and may well be higher than the limit assumed here. Additionally, repository size is fixed at 2,000 acres for the estimates in Table III-3. This limit is also site-specific.

With the exception of dome salt, little variation is seen in costs for design and construction. The higher cost in dome salt is attributable to possible increases in expenses for sinking shafts and constructing facility foundations in sites that may have complex hydrology and unconsolidated and saturated sediments. Current siting programs consider these possibilities in site evaluations with the intent of avoiding such conditions where possible.

Basalt and granite show clear increases in operating cost, which indicates the increased difficulty of excavation in hard rock media.

III.E.3.2.2 Thermal Loading Criteria

Thermal loading criteria affect repository capacity and therefore directly influence both repository total costs and undiscounted unit costs ($/kg). Many possible physical limits affect thermal loading criteria, and the controlling limit for each medium is still under study. Retrievability requirements can also strongly affect thermal loading criteria.

Table III-4 gives representative cost figures for basalt and granite repositories for each of two thermal loadings. Reduction in thermal loading can be seen to have a substantial effect on operating costs. This effect results from a decrease in total waste receipts that can be accommodated within a fixed repository size, thus decreasing expenditures for handling of waste and excavation. Undiscounted unit costs increase sharply, however, indicating that the total cost reductions achieved are not in proportion to the decrease in waste receipts.

Table III-3. Estimated Repository Costs for Various Geologic Media

	Standardized Repository			
Cost Item	Dome Salt	Bedded Salt	Basalt	Granite
Design and construction (M$)	1,210	950	930	920
Replacement capital equipment (M$)	50	50	100	100
Operating cost fixed (M$)	140	130	250	250
Operating cost variable (M$)	960	890	1,710	1,700
Operating cost total (M$)	1,100	1,020	1,960	1,950
Decommissioning cost (M$)	130	150	160	160
Total (M$)	2,490	2,170	3,150	3,130
Characteristics				
Heat loading kW/acre (design)	40	40	40	40
Canisters of spent fuel stored	160,000	160,000	160,000	160,000
MTU stored	68,000	68,000	68,000	68,000
Repository area (acres)	2,000	2,000	2,000	2,000
MTU/acre	34	34	34	34
Canisters/acre	80	80	80	80
Years of operation	14.9	14.9	14.9	14.9
Undiscounted unit cost ($/kg)	36.60	31.90	46.30	46.00

[a]All costs in 1980 dollars, unescalated, undiscounted.

III.E.3.2.3 Repository Size

The size of a repository site controls the ultimate capacity of the repository in terms of waste emplaced. For some types of sites, e.g., salt domes, size is limited by the nature of the specific geology. Since larger sites can receive more spent fuel, total operating costs will generally increase due to greater quantities of receipts and longer periods of operation. The converse of this statement is also true, unless size is sufficiently small to require the use of different techniques for excavation. Previous studies (47, 48, 57) have noted that a large site allows the development of mine layouts based on extremely long rooms that can be efficiently produced by high-production-rate mining equipment. Smaller sites are developed with

Table III-4. Estimated Repository Costs
for Varying Thermal Loadings

	Basalt		Granite	
Cost Item	100 kW/acre	40 kW/acre	100 kW/acre	40 kW/acre
Design and construction (M$)	930	930	920	920
Replacement capital equipment (M$)	120	100	120	100
Operating cost fixed (M$)	350	250	350	250
Operating cost variable (M$)	2,390	1,710	2,390	1,700
Operating cost total (M$)	2,740	1,960	2,740	1,950
Decommissioning cost (M$)	160	160	160	160
Total (M$)	3,950	3.150	3,940	3,130
Characteristics				
Canisters of spent fuel stored	400,000	160,000	400,000	160,000
MTU stored	170,000	68,000	170,000	68,000
Repository area (acres)	2,000	2,000	2,000	2,000
MTU/acre	85	34	85	34
Canisters/acre	200	80	200	80
Years of operation	32	14.9	32	14.9
Undiscounted unit cost ($/kg)	23.20	46.30	23.20	46.00

[a]All costs in 1980 dollars, unescalated, undiscounted.

shorter rooms, thus requiring more frequent setup times for mining equipment; consequently reductions in operating costs do not occur.

Table III-5 gives representative costs for three salt domes of differing size. Both effects discussed above can be seen in the results. Total cost decreases as dome size decreases from 2,800 acres to 2,000 acres, but then total operating costs remain constant in going from 2,000 acres to 1,200 acres because of the need for different mining approaches.

Table III-5 also shows that undiscounted unit costs decrease as size increases. This result is attributable to the fact that capital construction and decommissioning costs are relatively insensitive to repository size, and thus are prorated over a larger amount of receipts as the repository size increases.

Table III-5. Effects of Size on Salt Dome Repository Costs

	Standardized Repository		
Cost Item	1,200 Acre Repository	2,000 Acre Repository	2,800 Acre Repository
Design and construction (M$)	1,190	1,210	1,240
Replacement capital equipment (M$)	50	50	70
Operating cost fixed (M$)	140	140	170
Operating cost variable (M$)	960	960	1,240
Operating cost total (M$)	1,100	1,100	1,410
Decommissioning cost (M$)	130	130	130
Total (M$)	2,470	2,490	2,850
Characteristics			
Assumed heat loading kW/acre	40	40	40
Canisters of spent fuel stored	96,000	160,000	224,000
MTU stored	41,000	68,000	96,000
MTU/acre	34	34	34
Canisters/acre	80	80	80
Years of operation	10.4	14.9	19.5
Undiscounted unit costs ($/kg)	60.20	36.60	29.70

[a] All costs in 1980 dollars, unescalated, undiscounted.

III.E.3.3 Comparison with Previous Cost Data

For repositories receiving spent fuel in salt formations, total costs (design and construction, operations, and decommissioning) have been previously reported (34, 48, 57) that range from \$1.56 billion to \$1.92 billion (1978 dollars). Costs reported herein range from \$2.17 billion to \$2.85 billion, about 40% greater, on the average. The bulk of this increase is associated with escalation to current (1980) dollars. An increment of about 10% is attributable to the use in this discussion of thermal loadings lower than those generally used previously, and also the inclusion of cost data for small salt domes. Generally speaking, for constant dollar estimates, agreement is good for salt repositories, i.e. well within the stated accuracies.

For repositories in granite and basalt, previously reported costs (34) are \$4.96 billion and \$5.49 billion, respectively (1978 dollars). These are substantially larger costs than those discussed here. The principal difference is in the amount of excavation assumed. Estimates presented herein are based on standardized mine layouts including substantially less rock removal than those presented in EIS-0046-D (34). Excavation of these rock types is more expensive on a unit basis than excavation of salt.

III.E.4 Cost of Packaging/Encapsulation Facilities

A facility to encapsulate spent fuel has been conceptually designed (61). Additional studies of packaging/encapsulation facilities have also been made (34, 58). For the purposes of this discussion, the facilities described in the conceptual design document (61) are assumed to be located at the repository site. For the repositories listed in Table III-3, an estimate has been produced that scales packaging facilities to match the requirements of the repository. This estimate is summarized in Table III-6.

The facilities in the conceptual design report (61) prepare spent fuel packages that consist of intact spent fuel assemblies inside a mild steel, gas-filled canister. Subsequent research programs may indicate that more complex packaging may be required to achieve sufficient containment of radionuclides. Recent studies (62) have addressed a variety of potential package components and packaging processes. Substantial variations in packaging facility costs (perhaps as much as a factor of two) may be encountered in the event alternative package configurations are selected.

Table III-6. Representative Packaging/Encapsulation Facility Costs[a] for 2,000-Acre Repository Loaded With 40 kW/Acre

Cost Item	Cost ($Millions)
Capital line item cost (M$)	460
Capital equipment (M$)	70
Operating cost, fixed (M$)	360
Operating cost, variable (M$)	1,010
Operating cost, total (M$)	1,370
Decommissioning costs (M$)	40
Total (M$)	1,940
Characteristics	
Canisters processed	160,000
MTU processed	68,000
Years of operation	14.9
Undiscounted unit cost ($/kg)	28.52

[a]All costs in 1980 dollars, unescalated, undiscounted.

III.F SUMMARY OF SCHEDULE AND COST FACTORS FOR MINED GEOLOGIC DISPOSAL

III.F.1 Introduction

This chapter incorporates the conclusions of the preceding sections and presents an integrated overview of the schedule for a mined geologic repository and the level of confidence in proposed schedules and costs.

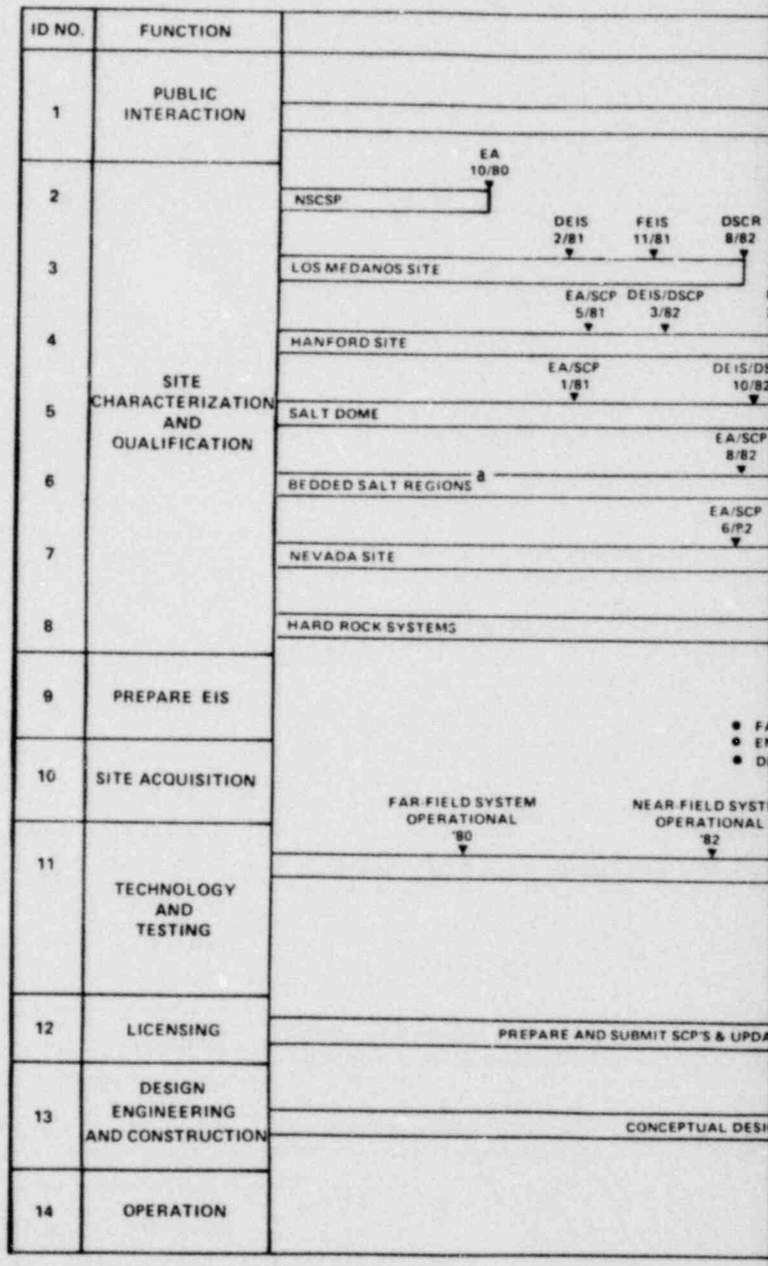
III.F.2 Factors

This section provides an integrated overview of the schedule for a geologic repository. Two schedules previously presented in Chapter III.C are repeated here. The reference schedule, Figure III-2, delineates those milestones deemed achievable in a reasonable time under present program policies. The extended schedule, Figure III-3, delineates those milestones that might occur if the amount of data required to support a license application increase and/or review periods were to become longer than expected. The second schedule also allows for delays in licensing and construction.

III.F.2.1 Integrated Time Line

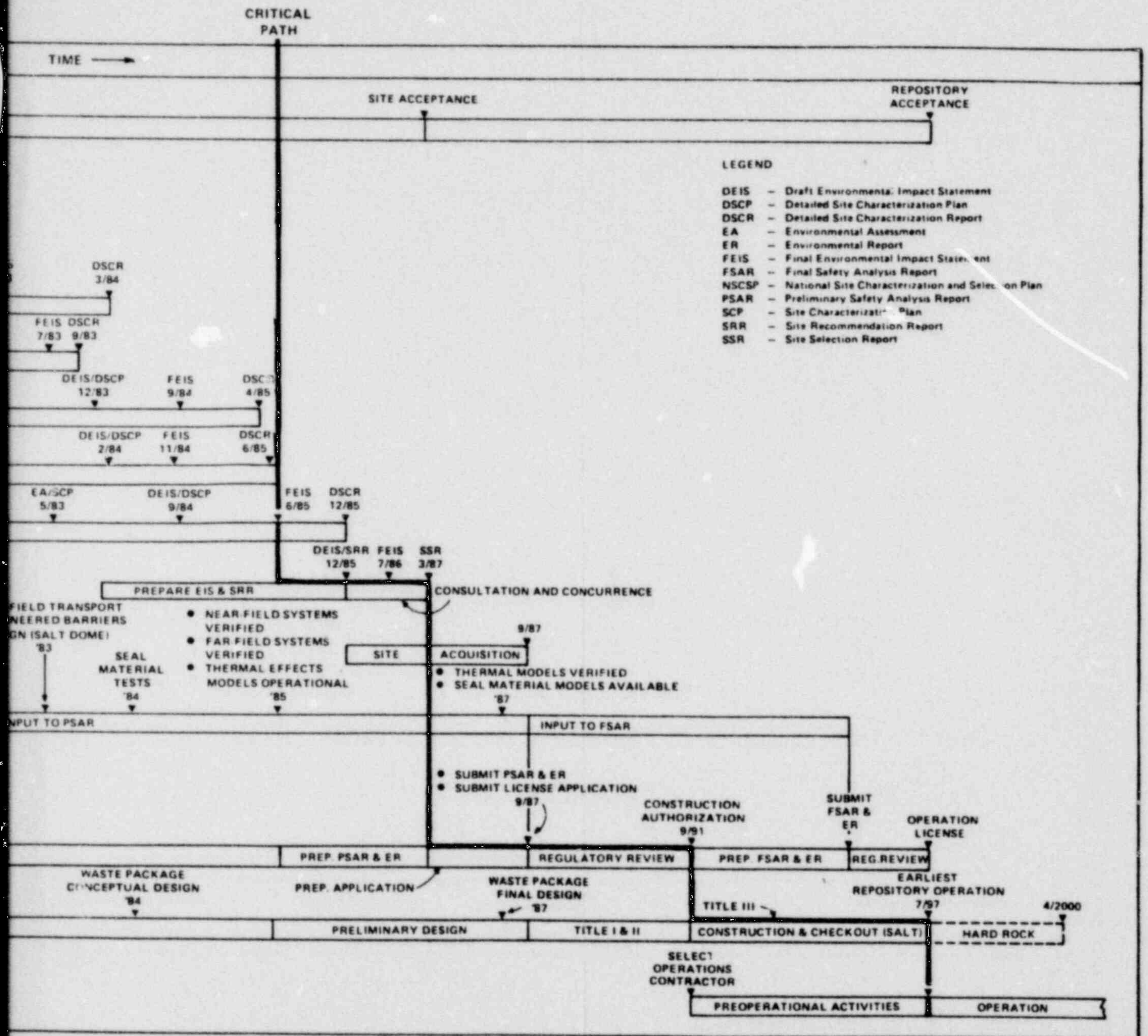
. Figure III-2 is an integrated time line diagram of the activities leading to the operation of a geologic repository. The critical path to operation of a geologic repository is illustrated by solid lines. The critical path follows those essential activities that must be performed in series and therefore define the time interval to the operational date. The critical path consists of the following activities, listed in order of their occurrence:

1. Multiple site characterization.
2. Site selection.
3. Licensing review to support a construction authorization.
4. Repository construction and checkout (preoperational tests).

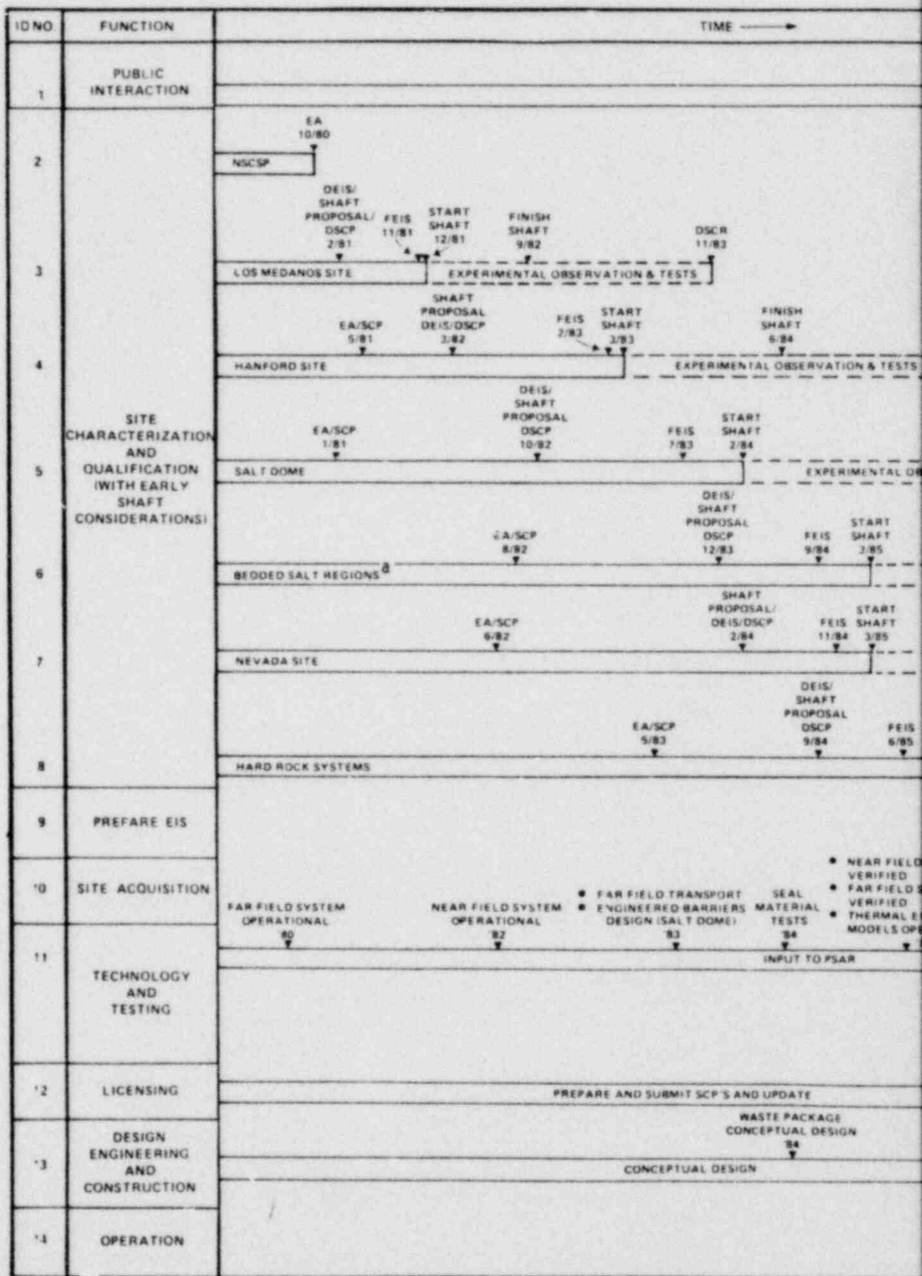


[a]Bedded salt regions other than Los Medanos.

Figure I



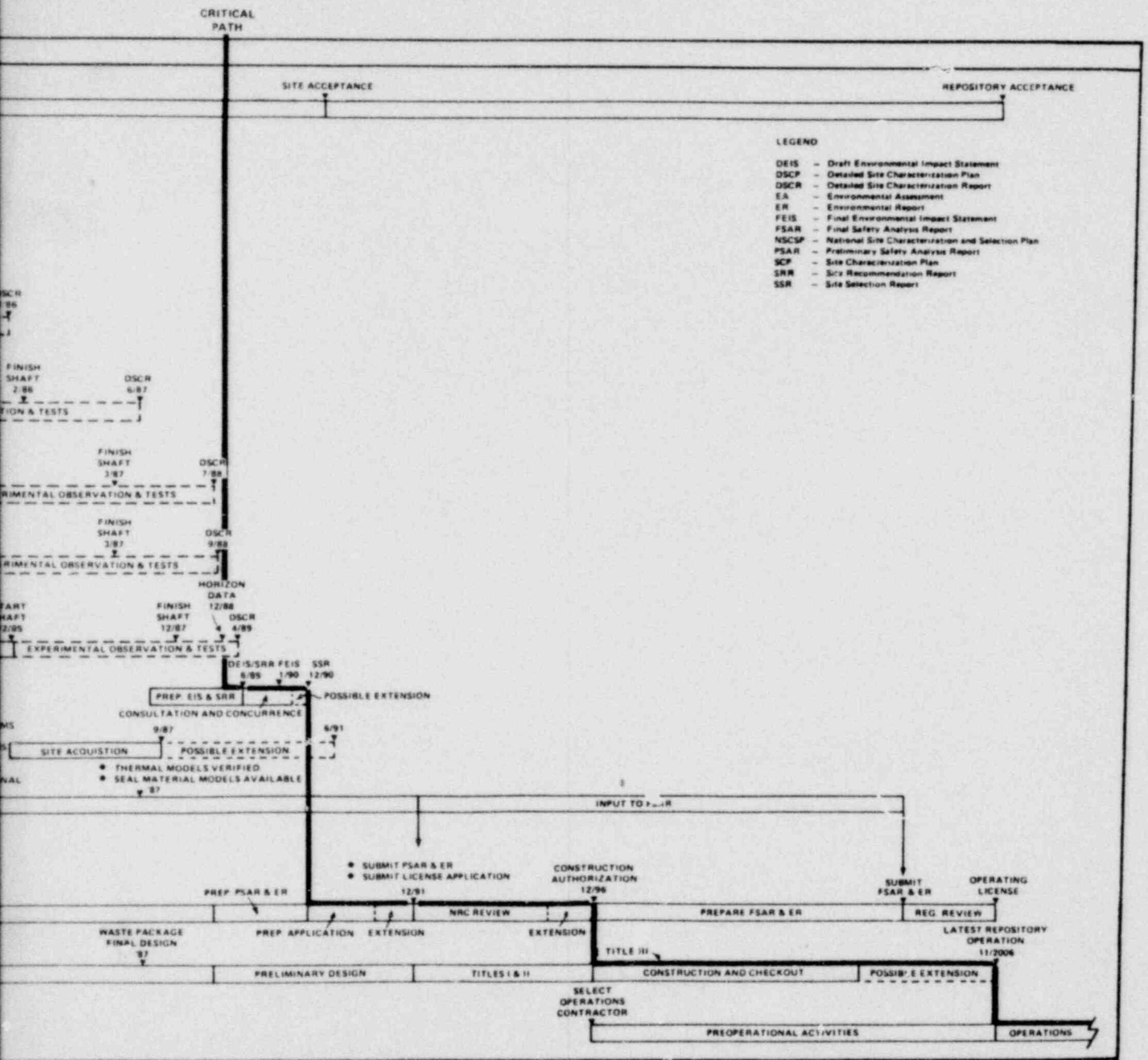
I-2. Summary Logic Network Activities Leading to Geologic Repository Operation (Reference Schedule)



[a]Bedded salt regions other than Los Medanos.

Figure II

(Extended dura



3. Summary Logic Network Activities Leading to
Geologic Repository Operation
on for characterization, licensing, and construction)

The duration of each activity is estimated using reasonably achievable goals. A range of durations for construction is presented to cover the variation in time required for construction of repositories in different media. The operational date is determined by summing these durations.

Current planning for the reference case is that the site recommendation report (SRR) for the initial repository will propose one of four or five previously banked sites for license application in December 1985 (see Figure III-2, line 9). If more site exploration and evaluation is required than is now anticipated, the extended case would assume that this milestone would be delayed an additional 40 months and would occur in June 1989 (see Figure III-3, line 9). Table III-7 lists the time estimates for the remaining activities.

Table III-7. Durations of Activities Following Issuance of Site Recommendation Report

	Duration, months	
Activity	Reference	Extended
(Site Recommendation Report Date)	(12/85)	(6/89)
Site Selection Decision	15	18
Application Preparation	6	12
Regulatory Review	48	60
Construction (depends upon mineral-type selected)	63-96	75-108
Checkout Tests	6	9
Total	138-171	174-207
(First repository operation date)	(1997-2000)	(2004-2006)

On the basis of these estimates and Site Recommendation Report dates of December 1985 and June 1989, a range of possible dates for operational startup of the first geologic repository is 1997-2006.

III.F.2.2 Technical and Social Aspects

Confidence in the schedule can be examined in view of technical considerations and social aspects.

III.F.2.2.1 Technical Considerations

The preceding section demonstrated that, with reasonable ranges of estimates for various activities in the schedule, a repository can be in operation between 1997-2000, depending upon the exact mineral type in which the repository is constructed. By considering possible delays in various steps in the process, it is further shown that such delays would not result in repository operation any later than 2004-2006; again, depending upon the exact mineral type chosen.

Chapter II.E concludes that anticipated technical problems can be resolved through carefully planned programs; application of the engineered barrier approach; comprehensive safety assessment modeling; and a program of field, laboratory, and in situ testing.

The present strategy for banking candidate sites provides for selection and licensing of proposed sites on a 3 year cycle, as needed. After a few years, multiple repositories will be under construction simultaneously. Therefore, if some unforeseen circumstance related to specific site, rock type, or technical concept that might cause abandonment of the site were to arise during construction or operation of the first repository, little delay would result, as other repositories would be available to receive waste in approximately 3 years. However, each such circumstance would be analyzed in detail and alternatives fully evaluated before such an action would be taken.

From these plans, approaches, and schedules it is evident that, from a technical standpoint, a repository can be constructed within a reasonable time.

III.F.2.2.2 Social Concerns

Because social concerns are less easily predicted, less confidence can be placed in assessment of their impacts on the repository program. Nonetheless, there is growing public recognition that nuclear waste management is a national problem and that solution of the problem should not be postponed for future generations. For example, the attention focused upon the need for additional low-level waste burial grounds, raised by the actions of authorities in the States of South Carolina, Washington, and Nevada, has resulted in active efforts by other States to address such problems. This developing national awareness also is reflected in the recent statement by the President which was based on the recommendations in the IRG report. The President confirmed the lead responsibility of the Department for coordinating waste management activities within the Federal structure. The result is a rapidly evolving nuclear waste disposal program with a broad scientific base. Included in the program are measures to allow for open interaction with the concerned public.

The continuing implementation of the Department's policy of consultation and concurrence and implementation of the Department's NEPA guidelines build confidence that the schedules provide for adequately addressing social concerns.

III.F.3 Ranges of Cost

The actual design, construction, and operating costs of any particular geologic repository will be based on a variety of factors, such as

1. Repository site conditions.
2. Repository media.
3. Repository size.
4. Nature of waste types received.
5. Operating period.

The range of variations in each of these factors could be substantial.

In past studies, the Department has considered the variations in these factors. Conceptual design activities have been conducted based on different fuel cycles and for differing site conditions. Feasibility studies (34) have examined the range of costs that accrue due to variations in geologic settings. Variations in repository size have been addressed in studies supporting the setting of spent fuel disposal charges to utilities. In addition, recent cost reconciliation studies (48) have allowed comparisons of differing estimates, with the conclusion that the ranges in estimates are reasonable and proper. Current estimates are considered to have an accuracy within 30%.

Deliberations of the International Nuclear Fuel Cycle Evaluation (INFCE) working group on nuclear waste management have concluded that the overall costs of waste management are small in relation to the total cost of electric power to the consumer, and are not sufficiently different between fuel cycles to allow preferential selection of one fuel cycle over another on economic grounds (63). The INFCE report was reviewed carefully by technical experts from several nations who participated in the INFCE Working Group on waste management.

The magnitude, breadth, and depth of cost analyses performed for conceptual geologic waste repositories, coupled with conservative scheduling, provide a basis for confidence in the range of anticipated costs for the first repository.

References for Part III

(1) Presidential Message to the Congress, February 12, 1980, "Comprehensive Radioactive Waste Management Program," Weekly Compilation of Presidential Documents, Vol. 16, No. 7

(2) Office of Nuclear Waste Isolation, NWTS Program Criteria for Geologic Disposal of Nuclear Waste--Site Qualification Criteria, ONWI 33-2, Battelle Memorial Institute, Columbus, OH, January 1980

(3) Office of Nuclear Waste Isolation, "NWTS Program Plan," ONWI Technical Program Plan, Vol. 1, Technical Overview, ONWI-19, Battelle Memorial Institute, Columbus, OH, February 1979

(4) Rockwell International, BWIP Program Plan, FY80, RHO-CD-132 Rev. 5, Rockwell Hanford Operations, Richland, WA, December 1979

(5) Nevada Nuclear Waste Storage Investigations, FY80 Project Plan, NVO-196-13, U.S. Department of Energy, Nevada Operations Office, Las Vegas, NV, 1980

(6) Office of Nuclear Waste Isolation, National Site Characterization and Selection Plan, Battelle Memorial Institute, Columbus, OH, in preparation

(7) Disposal of High-Level Radioactive Wastes in Geologic Repositories; Proposed Licensing Procedures, 10 C.F.R. 60; 44 Fed. Reg. 70408, December 1979

(8) National Environmental Policy Act of 1969, 42 U.S.C. 4321 et. seq.; amended by PL94-52, July 3, 1975; PL94-83, August 9, 1975

(9) Kenneth S. Johnson and Serge Gonzales, Salt Deposits in the United States and Regional Geologic Characteristics Important for Storage of Radioactive Waste, Y/OWI/SUB-7414/1, Earth Resource Associates, Inc., Athens, GA, March 1978

(10) Law Engineering Testing Company, Geologic Evaluation of Gulf Coast Salt Domes, National Waste Terminal Storage Program, ONWI LIB 0092 (Draft), Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, April 19, 1978

(11) K.S. Johnson, Evaluation of Permian Salt Deposits in the Texas Panhandle and Western Oklahoma for Underground Storage of Nuclear Wastes, Y/OWI/SUB-4494/1, Oak Ridge National Laboratory, Oak Ridge, TN, April 1976

(12) Office of Nuclear Waste Isolation, Regional Geology of the Salina Basin: Report of the Geologic Project Manager--Salina Basin, Phase 1, August 1977 - January 1978, Vol. I. New York and Ohio: Geology of Bedded Salt and Program Plan: Report of the Geologic Project Manager, Salina Basin, Phase 1, August 1977 - January 1978, Vol. II, ONWI/SUBE512-00600/1, Battelle Memorial Institute, Columbus, OH, January 1978. Possibilities for the Storage of Radioactive Waste in Salt Cavities in the Northeastern United States, prepared by the Geotechnical Corporation, Dallas, TX, February 1958

(13) R.J. Hite and S.W. Lohman, Geologic Appraisal of Paradox Basin Salt Deposits for Waste Emplacement, Open File Report 1973 USGS-4339/6, U.S. Geological Survey, 1973

(14) Rockwell International, Basalt Waste Isolation Project Annual Report FY1979, pp. I-11 to I-16, RHO-BWI-79-100, Rockwell Hanford Operations Energy Systems Group, Richland, WA, November 1979

(15) U.S. Department of Energy, Nevada Nuclear Waste Storage Investigation FY/80 Project Plan, NDO-196-13, Nevada Operations Office, Las Vegas, NV, March 1980

(16) Office of Nuclear Waste Isolation, Summary Characterization and Recommendation of Study Areas for the Gulf Interior Region, ONWI-18 (Final Draft), Battelle Memorial Institute, Columbus, OH, May 1979

(17) Office of Nuclear Waste Isolation, Summary Characterization and Recommendation of Study Areas for the Paradox Basin Region, ONWI-036 (Preliminary Draft), Battelle Memorial Institute, Columbus, OH, May 1979

(18) Office of Nuclear Waste Isolation, Preliminary Evaluation of the Environmental Aspects of Potential Radioactive Waste Repository Study Areas in the Ohio and New York Portions of the Salina Basin, ONWI-29, Battelle Memorial Institute, Columbus, OH, September 1979

(19) Office of Nuclear Waste Isolation, Regional Summary and Recommended Study Areas for Ohio and New York Portions of the Salina Basin, ONWI-29, Battelle Memorial Institute, Columbus, OH, September 1979

(20) U.S. Nuclear Regulatory Commission, Workshop Material for State Review of USNRC Site Suitability Criteria for High-Level Radioactive Waste Repositories, NUREG-0326

(21) Interagency Review Group Report to the President on Nuclear Waste Management, Washington, DC, TID-29442, March 1979

(22) Council on Environmental Quality, National Environmental Policy Act, Implementation of Procedural Provisions, Final Regulations, 43 Fed. Reg. 55978 (November 29, 1978); to be codified at 40 C.F.R., Part 1500 et seq.

(23) U.S. Nuclear Regulatory Commission, Means for Improving State Participation in the Siting, Licensing, and Development of Federal Nuclear Waste Facilities: A Report to the Congress, NUREG-0539

(24) "Consultation and Concurrence, Cooperative Federalism: A Reality for Successful Inter-Governmental Accomplishment," Proceedings of the Workshop on Consultation and Concurrence, East Sound, WA, September 23-26, 1979, ONWI-87, Battelle Memorial Institute, Columbus, OH

(25) Energy Reorganization Act of 1974, Secs. 202(3) and (4)

(26) Licensing Procedures for Geologic Repositories for High-Level Radioactive Wastes, 43 Fed. Reg. 53869 (November 1978)

(27) Joseph M. Hendrie, testimony before the Senate Committee on Environment and Public Works, September 1979

(28) U.S. Environmental Protection Agency, "Criteria for Radioactive Wastes," 43 Fed. Reg. 211 (November 1978)

(29) J.B. Martin, letter dated October 25, 1979, to C. Heath (DOE), "Comments on draft EIS (DOE/EIS-0046-D) by the NRC staff"

(30) W.N. Hedeman, Jr., letter dated September 27, 1979, to C. Heath (DOE), "Comments on draft EIS (DOE/EIS-0046-D) by EPA"

(31) Footnote to proposed 10 C.F.R. 51.40 "Environmental Reports"; 44 Fed. Reg. 70415 (December 1979)

(32) Proposed DOE Guidelines for Compliance with the National Environmental Policy Act 44, Fed. Reg. 42135 (July 18, 1979)

(33) 40 C.F.R. 1506.1

(34) U.S. Department of Energy, Management of Commercially Generated Radioactive Waste, Draft Environmental Impact Statement, Vols. 1 and 2, DOE/EIS-0046-D, April 1979

(35) Interagency Agreement (DOE/USGS), No. DE-A197-79ET46611, May 1, 1979; Mod. 2: October 1, 1979

(36) N.M. Fenneman, "Physiographic Divisions of the United States," Annnals of the Association of American Geographers, Third Edition, Vol. 18, pp. 261-353, 1928

(37) Cooperative Agreement Between U.S. Department of Energy and Bureau of Land Management on Site Investigation and Drilling at Los Medanos (WIPP) Site, EW-78-A-04-4185

(38) Memorandum of Agreement (Corps of Engineers), ED-A197-79ET46600, March 1, 1979; Mod. 1: October 1, 1979

(39) Memorandum of Agreement (Corps of Engineers), ED-A197-79ET46612, September 24, 1979; Mod. 1: December 20, 1979

(40) Formal Memorandum of Understanding, United States Department of Agriculture, Forest Service, DeSoto National Forest, MS, February 12, 1979

(41) 42 U.S.C. 7257

(42) DOE Notice 4300.1 of 9 February 1979 as extended by DOE Notice 1321.77 of 12 February 1980

(43) 41 C.F.R. 101-47.203 (1979)

(44) Federal Land Policy and Management Act, 43 U.S.C. 1701 et seq.

(45) 43 U.S.C. 1714(c)

(46) 44 Fed. Reg. 69868 (December 4, 1979)

(47) Stearns-Roger Engineering Company, Conceptual Design Report, National Waste Terminal Storage Repository for Storing Reprocessing Wastes in a Dome Salt Formation, DOE Contract No. EY-77-C-05-5367, U.S. Department of Energy, January 1979

(48) Kaiser Engineers, Inc., Conceptual Design Report, National Waste Terminal Storage Repository in a Bedded Salt Formation for Spent Unreprocessed Fuel, DOE Contract No. EY-77-C-05-5366, U.S. Department of Energy, December 1978

(49) Office of Waste Isolation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-36/I-23, Oak Ridge, TN, April 1978

(50) Assistant Secretary for Energy Technology, Energy Systems Acquisition Project Plan, U.S. Department of Energy, July 12, 1979

(51) U.S. Energy Research and Development Administration, Alternatives for Managing Wastes for Reactors and Post-Fission Operations in the LWR Fuel Cycle, Vol. 4 ERDA-76-43, May 1976

(52) Stearns-Roger Engineering Company, Special Study 3: Waste Retrieval from Backfilled Regions, DOE Contract No. EY-77-C-05-5367, U.S. Department of Energy, January 1979

(53) Kaiser Engineers, Special Study 3: Waste Retrieval from Backfilled Regions, DOE Contract No. EY-77-C-05-5366, U.S. Department of Energy, September 1978

(54) Office of Nuclear Waste Isolation, Waste Package Design, Development, and Test Plan, ONWI Draft, Battelle Memorial Institute, Columbus, OH

(55) Office of Nuclear Waste Management and U.S. Geological Survey, Earth Science Technical Plan for Mined Geologic Disposal of Radioactive Waste, Battelle Memorial Institute, Columbus, OH, April 1980

(56) Assistant Secretary for Nuclear Energy, U.S. Department of Energy, Nuclear Waste Program Summary-FY81, January 1980

(57) Stearns-Roger Engineering Company and Kaiser Engineers, National Waste Terminal Storage Repository 1 and 2 Cost Estimate Reconciliation Study: Vol. I, Phase I Report, August 1979; Vol. II, Phase II Report, October 1979.

(58) Bechtel National, Inc., An Assessment of LWR Spent Fuel Disposal Options, Vols. 1 and 2, ONWI-39 (Draft Final Report), Battelle Memorial Institute, Columbus, OH, April 1979

(59) Office of Waste Isolation, ...nical Support of GEIS: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-36/1-23, Oak Ridge, TN, April 1978

(60) U.S. Department of Energy, Preliminary Estimates of the Charge for Spent Fuel Storage and Disposal Services, DOE/ET-0055, July 1978

(61) Rockwell Hanford Operations and Kaiser Engineers, Conceptual Design Report, Spent Fuel Receiving and Packaging Facility, DOE Contract EY-77,C-06-1030, RHO-CD-506, Richland, WA, September 1978

(62) D.H. Laster et al., Systems Study on Engineered Barriers, Task 4, SAI Rep. 01379-990 LJ, ONWI/SUB/79E1515-02000-1, Battelle Memorial Institute, Columbus, OH, September 1979

(63) Waste Management and Disposal Report of Working Group, INFCE PC/2/7 PC/2/7, January 1980

IV TECHNICAL BASIS FOR CONFIDENCE THAT SPENT FUEL CAN BE STORED IN A SAFE AND ENVIRONMENTALLY ACCEPTABLE MANNER UNTIL DISPOSAL FACILITIES ARE AVAILABLE

Previous parts have discussed the ultimate disposal of spent fuel in geologic repositories and have forecast the probable time of availability of these facilities. Until that time, spent fuel will be stored at reactor facilities and, to the extent necessary, at away-from-reactor (AFR) facilities. This part discusses the technical basis which establishes that spent fuel can be stored until disposal facilities are available. Considered herein are the performance requirements for spent fuel storage facilities, the options available for storage, how safe and environmentally acceptable storage will be achieved, and the experience which supports the conclusion that extended storage can be achieved in a safe and environmentally acceptable manner.

This part first outlines the generic performance requirements for the storage of spent fuel (IV.A). These requirements have been derived largely from existing regulatory criteria. Using these requirements as benchmarks, the statement examines water pool storage and the various methods of dry storage and concludes that at the present time, water pool storage should remain the preferred method of storage, though dry storage is being demonstrated and appears to be a technically viable alternative (IV.B). Upon concluding that water pool storage is the preferred method of storage, the specific requirements for water pool storage are discussed. These requirements are primarily based upon regulatory and industry-developed standards (IV.C).

This part also demonstrates that safe and environmentally acceptable extended storage can be achieved (IV.D). Water pool storage associated with nuclear facilities has been licensed in the past, and the technology is available to meet regulatory requirements. It is concluded, in the words of the Nuclear Regulatory Commission's final GEIS on Handling and Storage of Spent LWR Fuel, that (1)

> The storage of LWR spent fuel in water pools has an insignificant impact on the environment, whether such pools are at reactor sites or away therefrom . . . The technology of water pool storage is well developed. . . radioactive waste that is generated is readily confined and presents little potential hazard to the health and safety of the public.

IV.A HIGH-LEVEL WASTE STORAGE SYSTEM OBJECTIVES

The primary objective of spent-fuel management is that spent fuel be stored in a safe, economical, and environmentally acceptable manner until it is transferred to a repository for ultimate disposal. This may be accomplished in storage facilities at the reactor, in storage facilities located away from the reactor, or by a combination thereof. The extensive prior spent-fuel storage experience, monitoring programs to confirm the continuing integrity of fuel in storage, and the development and availability of additional storage options, demonstrate that interim storage can be provided for as long as may be necessary.

The goals for a successful spent-fuel storage program may be expressed in terms of the performance requirements which must be met by the storage facilities and the period of time during which such storage must be available. These are discussed in the following paragraphs.

IV.A.1 Generic Performance Requirements for Safe and Environmentally Acceptable Storage of Spent Fuel

A spent-fuel storage facility must be constructed and operated in such a manner to ensure that certain generic performance requirements are met. These requirements follow from regulations of the Nuclear Regulatory Commission (NRC) which pertain to nuclear facilities, and they are amplified in a number of regulatory guides and industry standards. Most of the NRC requirements have been promulgated in final rules. The regulatory requirements relating to storage are thus more fully developed than those for disposal. The principal considerations are summarized in the following paragraphs.

IV.A.1.1 Release Limits

The release of radionuclides to the environment during the course of normal operations (in airborne, liquid effluent, and contamination forms) must be maintained below the limits established in 10 CFR 20 (2) for nuclear operations. 10 CFR 20 sets forth the Nuclear Regulatory Commission standards for protection against radiation. It specifies allowable contamination levels in facilities and in the environment around the site, allowable release levels for radioactive materials from the facility, and allowable radiation exposures to facility employees and the public as a result of facility activities. Supplementing the specific values set forth in 10 CFR 20, 10 CFR 20.1 requires licensees to limit releases and exposures to "as low as reasonably achievable"--the ALARA standard.

IV.A.1.2 Radiation Exposure

Exposure of individuals to radiation must be maintained below the limits set forth in 10 CFR 20 (2).

IV.A.1.3 Accidents

The facility must be designed to minimize the possibility of the following types of occurrences during the operation of spent-fuel storage facilities:

1. Criticality.
2. Loss of shielding and/or complete loss of cooling capability.
3. Damage to stored fuel caused by the dropping of heavy objects.
4. Multiple massive ruptures to spent fuel by tornado missiles.

Safety features must be provided to limit the consequences in the remote event that there should be such an occurrence.

The release of radionuclides to the environment during credible accident or abnormal operation sequences, and during the course of catastrophic events of natural phenomena, must be maintained below the guideline values set forth in 10 CFR 100 (3). (See II.F.2.5.) Nuclear Regulatory Commission regulation 10 CFR 100 prescribes the criteria for evaluation of prospective sites for nuclear facilities, among which are guideline values for releases of radioactive materials during accident situations at operating facilities. Dose guidelines for the accident situation are also specified in the Commission's proposed regulation (10 CFR 72.67(b)) covering independent fuel storage installations (4). If adopted as presently written, the guideline value of 10 CFR 72.67(b) will impose a substantially more conservative requirement on independent spent-fuel storage installations than is imposed on reactor and fuel reprocessing facilities.

IV.A.2 Period of Storage

The Department has concluded that a repository should be available for use in the 1997-2006 time period, depending on the geologic media selected (see Figures III-2, III-3). Even with these prospective dates of availability of a spent-fuel repository, it is prudent to design and construct spent-fuel storage facilities such that they can provide safe storage for a period of 40 years or more. Although experience with the water pool storage mode indicates that longer periods of safe storage are feasible (see IV.D.4), the 40-year goal for AFR storage capability is suggested on the bases that (i) it would be expected that spent fuel accumulated in reactor pools would, to the maximum extent practicable, be transferred directly to the repository rather than from reactor to AFR to repository; (ii) projected reception rates for the repository indicate that transfer of the backlog of spent fuel from reactors can be expected to take place over a number of years after activation of the repository; and (iii) 40-year storage capability provides the flexibility to deal with unanticipated delays in transfers of fuel to the repository.

IV.B ALTERNATIVES AND PREFERRED METHOD FOR STORAGE

There are two basic methods available for storage of spent fuel--water pool storage and dry storage, which may be provided in more than one way. The Department and most utilities have concluded that water pool storage is the preferred method of interim storage; and, in fact, water pool facilities must be used by utilities to receive fuel freshly discharged from light-water reactors because of the high thermal output of these assemblies (see Table IV-1). This section briefly describes the various types of storage methods available and explains the rationale behind the choice of water pool storage as the preferred method.

Table IV-1. Representative Thermal Power and Radioactivity Levels of Spent Nuclear Fuel From a Light Water Reactor

Time from Discharge from Reactor (years)[a]	Approximate Thermal Power of Spent Fuel[b] (watts/MTU)	Approximate Radioactivity Levels of Spent Fuel (Ci/MTU)
At Discharge	1,700,000	1.43×10^8
0.5	16,700	4.2×10^6
1	10,200	2.5×10^6
10	1,200	4.0×10^5
30	720	2.0×10^5

[a]Pressurized water reactor, burnup--33,000 MWd/MTU.

[b]MTU = metric tons of uranium in the unirradiated fuel charged to the reactor.

Source: Adapted from Table II-4 in Chapter II.C.

In any method of spent-fuel storage, the basic functions of the facilities required are these:

Removal of heat from the fuel--to carry away the heat produced by radioactive decay of the fission products and transuranium elements (such as plutonium and americium).

Radiation shielding--to maintain acceptably low levels of radiation in working areas.

Containment of radioactive materials--to prevent the escape of radioactive materials into working areas and the general environment.

Heat removal is effected by use of a cooling medium, such as water or air. Radiation shielding is accomplished by storing the fuel under an appropriate depth of water or by containing the fuel in a cavity surrounded by solid material (such as concrete or earth) sufficiently thick to prevent significant levels of radiation from penetrating it. Containment is accomplished by storing the fuel under water, in a vault, or in a container which prevents the release of significant quantities of radioactive material even during credible accident situations, supplemented, as necessary, by facilities for removal and recovery of any radioactive materials that escape into the cooling medium.

Although the generation of both heat and radioactivity is continuous, the amount of heat produced and the amount of radiation generated decrease with time as the decay of the fission products proceeds. This is demonstrated by the data set forth in Table IV-1.

Although both the heat and radiation levels diminish with time, it is necessary to provide cooling capacity and shielding capable of accommodating the spent fuel when it is initially received into the storage facility. In later years of storage, the cooling capacity is considerably in excess of that actually needed, but radioactivity levels are still high enough within the time frame considered likely for interim storage to require a level of shielding similar to that required initially. (Although the radioactivity levels decline with time at about the same rate as the thermal output, the drop in shielding requirement is not linear.) It is expected that most of the spent fuel which is destined for storage in an AFR storage facility will have been in the reactor storage pool for at least 5 years prior to transfer to the AFR storage facility.

IV.B.1 Alternative Storage Methods

Spent fuel can be stored by either of two basic methods--wet storage or dry storage. The following paragraphs describe the alternatives available for spent-fuel storage and the method currently preferred for spent light water reactor (LWR) fuel.

IV.B.1.1 Water Pool Storage

IV.B.1.1.1 Description of Water Pool Storage

Water pool storage consists of storing spent fuel in racks generally positioned at the base of a pool of water. The water serves as a heat transfer agent to remove the heat from the stored fuel and also provides radiation shielding; the water depth above the stored fuel is sufficient to prevent the escape of any significant amount of radiation therefrom (about 11-13 ft of water above the top of the stored assemblies). The principal source of radioactive contamination in reactor storage pools comes from mixing of reactor coolant with storage pool water during spent fuel transfers. Some contamination is on the fuel in the form of activated corrosion products from the reactor primary coolant system. In the storage pool, some radioactive contamination (primarily cesium, cobalt, and strontium) of the pool water results from the leaching of this surface contamination (called "crud") from the assemblies and from leakage from damaged fuel assemblies, although the latter source has been found to be a very minor one (see IV.D.4.2). Buildup of radioactivity levels in the pool water is prevented by continuously withdrawing a portion of the water and circulating it through a purification system which removes the radioactive contamination by a combination of filtering and ion exchange. The heat which has been added to the water by radioactive decay in the fuel is removed by passing this same sidestream of pool water through a cooler ("heat exchanger") before returning it to the pool. The purpose of this treatment is twofold: to ensure that high water quality is maintained in the pool and to maintain the temperature of the pool at prescribed levels.

The water pool used for storage generally consists of a thick reinforced concrete structure that is lined with stainless steel. The spent fuel assemblies are stored by positioning them in basket-type canisters in storage racks which are mounted to the base of the pool. The racks provide for a positive spacing between stored fuel assemblies to prevent criticality. If stainless steel is used as the basket or rack material, spacing can be relatively close, as stainless steel is a neutron absorber. Additional reduction in spacing can be achieved by introducing a stronger neutron absorber such as boron in the structure of the racks or storage baskets while maintaining the required margins of criticality safety.

If damaged fuel is received for storage, or if fuel should lose its containment integrity through damage during storage, such fuel may be encapsulated in a metal container and the container stored in the manner described.

Figure IV-1 diagrams the basic operations involved in water pool storage of spent fuel; the arrows illustrate the movement of elements within the system.

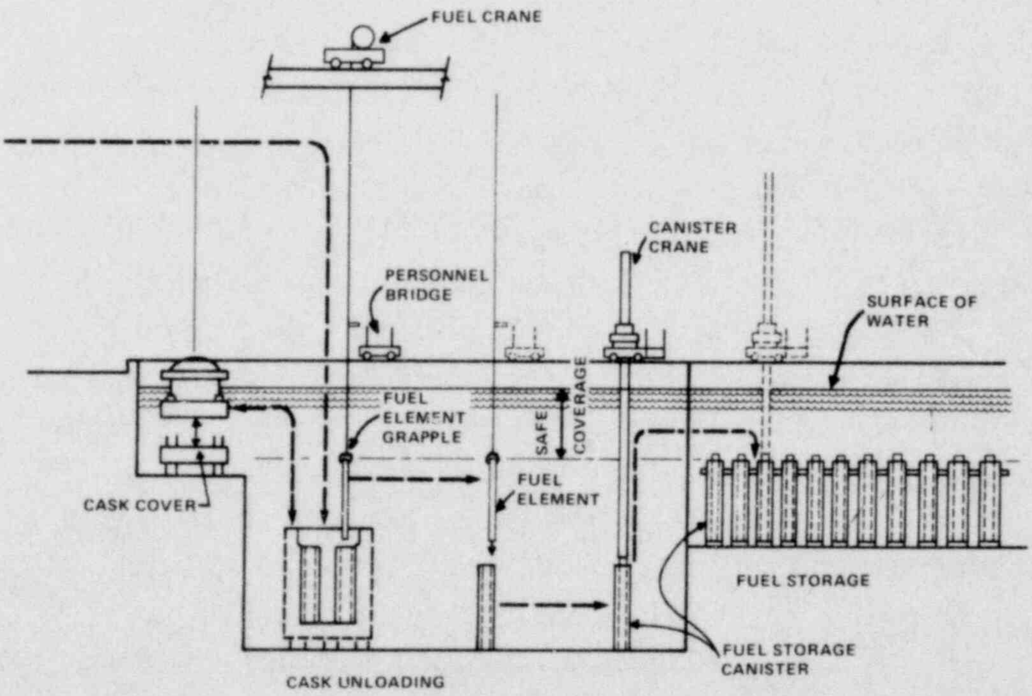


Figure IV-1. Basic Operations of Water Pool Storage of Spent Nuclear Fuels

In addition to the basic features of a water pool storage facility, as described above, the storage facility generally provides for the following operations and services:

1. Cask washdown, cooling, unloading, and decontamination.
2. Radioactive waste processing and storage.
3. Utilities and services.
4. Heating and ventilating.
5. Maintenance.
6. Fuel encapsulation.
7. Decontamination.
8. Health physics.
9. Laboratory.
10. Personnel and administration.

In cases where the storage facility is colocated with other nuclear activities (such as a nuclear power plant), some of the listed auxiliary and support facilities will be common to the other operations at the site.

IV.B.1.1.2 Experience with Water Pool Storage

Water pool storage of spent fuel (and other highly radioactive items) has been practiced in the United States for more than 30 years; the knowledge gained as a result of this experience is discussed in IV.D.4. The knowledge and experience gained in the operation of fuel storage pools have been utilized both by the Department of Energy and industry in a number of different designs for water pool facilities of varying capacities, ranging from a few hundred tons to 10,000 tons of spent fuel (5-9). Modular design concepts for AFR storage pools have been developed which permit addition of capacity to a basic pool in increments of a few hundred tons of spent fuel (10).

All existing and planned water-cooled reactors, both United States and foreign, have provisions at the reactor site for the water pool storage of spent fuel. Table IV-2 sets forth the number of power stations that have provisions for water pool storage (11). The data in Table IV-2 testify to the general acceptance of this method of spent fuel storage and to the scope of experience which is accumulating. Although the operational concept of at-reactor storage of spent fuel initially envisioned relatively short-term (i.e., 6-12 months) storage at the reactor, during the past 7 years increasing attention has been given to the prospects of much longer storage times, as a result of uncertainties in the "back end" (closing) of the fuel cycle.

A large number of the United States nuclear power reactors have for this reason either increased their pool capacity or are now in the process of expansion (12-14). Increase in pool capacity at existing reactors is most frequently accomplished by increasing the storage density; this may be done by replacing aluminum racks with stainless steel or by adding neutron-absorbing curtains between fuel positions to reduce neutron interaction. As more precise nuclear criticality computations have become available, it has been possible to design for somewhat closer spacing than was permitted in the original installations, thus affording some gain in capacity. In some instances, additional capacity has been achieved by installation of storage racks in portions of the storage pool which were not previously occupied by such racks. Reactors in design or under construction are expanding the size and capacity of their spent fuel pools where such can conveniently be done.

There has also been significant experience developed in regard to pool storage of DOE and commercial spent-fuel assemblies at reprocessing sites. Typical of United States off-reactor site pool storage are the operations shown in Table IV-3.

In addition, there are a number of foreign reprocessing facilities which have a spent-fuel storage capability including, but not limited to, those in the United Kingdom, France, West Germany, Belgium, Italy, Japan, India, and Canada. Zircaloy-clad spent fuel from a Canadian test reactor (NPD) has been stored in a water pool for 16 years at the Chalk River Nuclear Laboratories (15).

Table IV-2. Reactors with Water Storage Pools for Spent Fuel Storage

Nuclear Power Stations	Number in United States	Number of Foreign
Licensed for commercial operation[a]	70	138[b]
Under construction	90[c,d]	166[d]

[a]Includes two reactors not formally licensed and three which are shut down at present.

[b]These figures include some gas cooled reactors and advanced gas reactors that use water pool storage.

[c]An additional 27 reactors are on order in the United States, but not yet under construction.

[d]Does not include reactors with indefinite schedules.

Source: (Reference 11) American Nuclear Society, "World List of Nuclear Power Plants," Nuclear News, 23, No. 2, pp. 67 ff, February 1980.

Table IV-3. Spent Fuel Storage Experience in Water Pools at Storage Facilities in the United States

Facility	Operation	From	To
Hanford Works (DOE)	Storage of DOE spent fuel	1945	Present
Idaho National Engineering Laboratory (INEL):			
Chemical Processing Plant	Storage of DOE spent fuel	1953	Present
Expended Core Facility	Storage and examination of DOE spent fuel	1959	Present
Savannah River Plant (DOE)	Storage of DOE spent fuel	1953	Present
NFS-West Valley	Storage of commercial spent fuel	1966	Present
MFRP-GE-Morris	Storage of commercial spent fuel	1972	Present

IV.B.1.2 Dry Storage in Vaults

IV.B.1.2.1 Description of Dry Vault Storace

Dry storage of spent fuel in vaults consists of storing the spent fuel in racks or on hangers, in air, inside a concrete vault structure or storage in cavities in a monolithic concrete structure. Forced-air circulation or passive natural draft is used to cool the stored fuel while the reinforced concrete structures provide the necessary radiation shielding. Cooling air is filtered through high efficiency filters and is monitored prior to release to the atmosphere. While spent fuel could be stored in either the encapsulated or unencapsulated form, the former has been favored because of the reduced air treatment facilities and operations that are required.

Figure IV-2 diagrams the basic operations involved in dry vault storage of spent fuel.

In addition to the basic features of the storage system described above, the storage facility generally has the same types of auxiliary and support facilities described for water pool storage except that the fuel encapsulation facilities may be more extensive.

IV.B.1.2.2 Experience with Dry Vault Storage

In the United States, experience with dry vault storage includes the handling and storage of graphite-based fuel from nuclear rocket research and development programs, which was temporarily placed in dry storage at the Nevada Test Site before transfer to a specially designed dry storage vault at the Department of Energy's Idaho Nuclear Engineering Laboratory (INEL), and the storage of high-temperature, gas-cooled reactor (HTGR) fuel at reactor sites. Though the former fuel is very low exposure fuel which does not present the same heat removal problems as would higher exposure power reactor fuel, the dry handling and storing techniques have been demonstrated.

The dry vault storage experience gained has not involved spent fuel that is typical of that discharged from commercial light-water reactors; still, the basic principles of heat transfer and shielding in dry storage have been demonstrated during these activities and provide important basic design information which would be necessary for the design and construction of such dry storage facilities for spent LWR fuel. LWR fuels would, in fact, be more easily handled in dry storage than the graphite fuels, which require the exclusion of moisture. Typical of U.S. operations involving dry storage of spent fuel are those in Table IV-4.

Moreover, the proposed HTGR plants designated as the Fulton Station (16) and the Summit Station (17) were designed for dry storage of spent fuel; and nine gas-cooled reactor stations are utilizing dry storage of spent fuel in foreign countries. This indicates the availability of adequate engineering design information for such facilities.

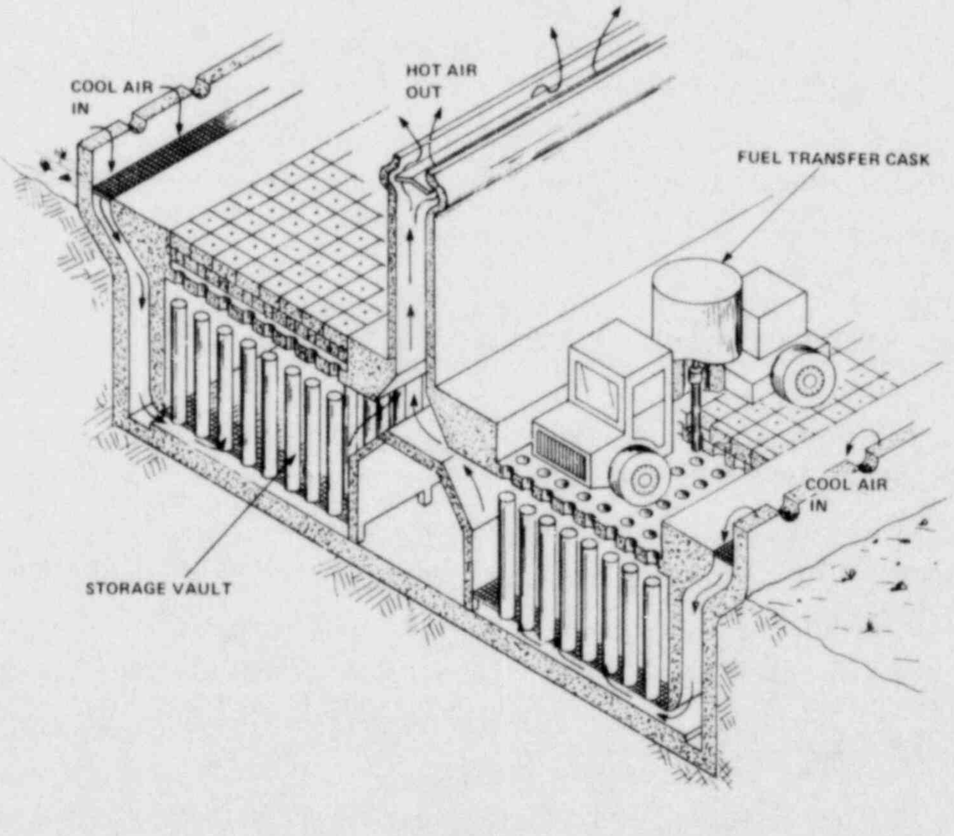


Figure IV-2. Basic Operations of Dry Vault Storage of Spent Fuel

Table IV-4. Spent Fuel Storage Experience With Dry Vault Storage in the United States

Facility	Operation	From	To
Peach Bottom 1[a]	Power reactor	1967	1974
Idaho National Engineering Laboratory[b]	Spent fuel storage--GCR and Rover/NERVA fuel	1969	Present
Fort St. Vrain[c]	Power reactor	1977	Present
Engine Maintenance and Disassembly Facility (EMAD)[d]	Research and development	1979	Present

Sources:
[a](Reference 18) U.S. Nuclear Regulatory Commission, Facilities License Application Record, License Docket 50-171

[b](Reference 19) U.S. Energy Research & Development Administration, Waste Management Operations, Idaho National Engineering Laboratory, ERDA-1536, September 1977

(Reference 20) W. Hammond, R. P'Pool, and R.D. Modrow, Safety Analysis, Peach Bottom Spent Fuel Storage IN-1465, Idaho National Engineering Laboratory, Idaho Falls, ID

(Reference 21) U.S. Energy Research and Development Administration, Environmental Impact Statement-Receipt, Storage, and Processing of Rover Fuel, WASH 1512, April 1972

[c](Reference 22) U.S. Nuclear Regulatory Commission, Facilities License Application Record, License Docket 50-267

(Reference 23) U.S. Energy Research and Development Administration, Environmental Statement-HTGR Fuels Reprocessing Facilities, National Reactor Testing Station, WASH-1534, January 1974

[d](Reference 24) R.J. Steffen, D. Ourrill, and J.B. Wright, "EMAD Support of NWTS Experiments and Demonstrations", Proceedings of the National Waste Terminal Storage Program Information Meeting, ONWI-62, Battelle Memorial Institute), Columbus, OH, October 1979

IV.B.1.3 Dry Storage in Caissons Below Grade

IV.B.1.3.1 Description of Caisson Storage

Another method for storing of spent fuel under development and study consists of storing encapsulated fuel in suitably designed caissons. The caissons consist of steel sleeves with a grout plug at the base, which are buried in the ground and are sealed with thick concrete plugs. The spent fuel is encapsulated in a steel container, the encapsulated fuel is placed in the caisson, and the plug is inserted to seal the capsule therein. The depth from the surface to the top of the fuel is 8-10 ft. The caissons are located sufficiently far apart to eliminate the need for any active cooling system, inasmuch as the spacing allows for heat removal to the air via the surrounding ground. The concrete plug and the caisson/ground combination provide the necessary radiation shielding. Figure IV-3 diagrams the storage of spent fuel in a below-ground caisson.

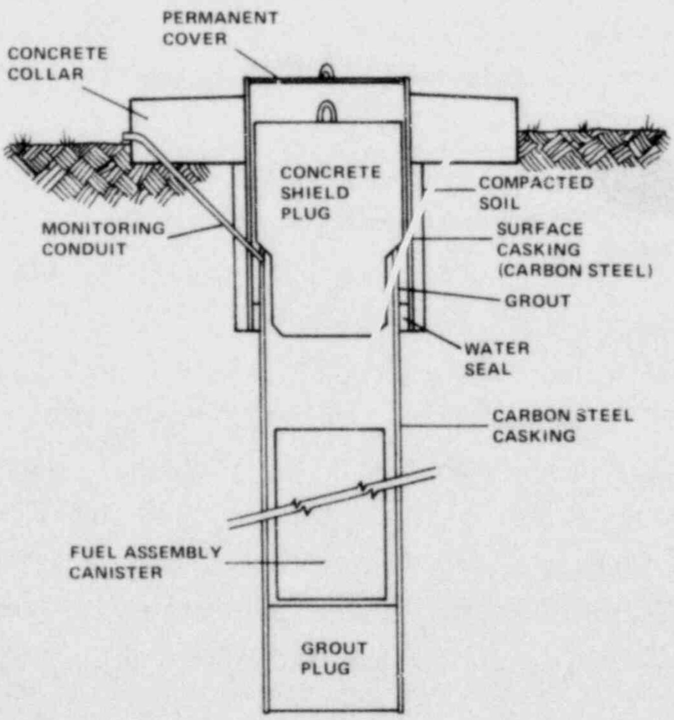


Figure IV-3. Below-Ground Caisson Storage of Spent Fuel

In addition to the caisson storage system, the storage facility generally has the same types of auxiliary and support facilities as those described for water pool storage except that (i) the fuel encapsulation facilities are more extensive, inasmuch as it is necessary to encapsulate all spent fuel prior to storage, and (ii) caisson and canister maintenance facilities are required.

IV.B.1.3.2 Experience with Caisson Storage

Storage of spent fuel in caissons below grade is undergoing tests by the Canadians and has been used on a small scale for interim storage of Peach Bottom Reactor fuel at INEL since 1970. A number of spent-fuel assemblies from the Peach Bottom reactor have been stored in 30-in. carbon steel caissons in the desert soil (20). These fuel assemblies, irradiated to greater than 26,000 MWd/MTU and 3-5 years out of the reactor, are contained in aluminum canisters. Data confirming design calculations have been obtained. Results of the INEL work and of calculations made by Atlantic Richfield Company (25) in the United States, and by Atomic Energy of Canada, Limited (AECL) in Canada (26), indicate that this method of interim storage of spent fuel is feasible.

In early 1979, a program was initiated at the Engine, Maintenance, Assembly, and Disassembly facility at the Nevada Test Site (NTS) on the encapsulation and caisson storage of spent LWR fuel, and this program is continuing (24).

This program is part of an overall effort under DOE sponsorship to demonstrate the technology of handling and encapsulating spent fuel, to obtain design data on spent fuel capsules, storage caissons, and surface storage casks, and finally, to provide encapsulated spent fuel for use in the geologic storage test program at the NTS Climax Granite Stock. Three PWR assemblies were encapsulated and placed in storage in late 1978 and 1979, and 13 more are being encapsulated for the geologic storage tests.

IV.B.1.4 Dry Storage in Concrete Storage Casks Above Grade

IB.B.1.4.1 Description of Concrete Cask Storage

The concrete cask storage concept differs from both the water pool concept and the air-cooled vault concept in that no storage vault is required; instead, each spent fuel canister is stored outdoors within its own massive cask and concrete shield, which provide both structural ruggedness for contamination confinement under all credible accident conditions and natural forces, and radiation shielding. The concept is similar to the air-cooled vault concept in that the radioactive decay heat is dissipated by passive (natural draft) air cooling, or by radiation of the heat from the outer surface of the concrete cask.

The spent fuel is encapsulated in a heavy steel container; the canister for storage by radiation cooling typically holds one pressurized water reactor (PWR) or three boiling water reactor (BWR) assemblies, and the canister for storage by natural convection cooling typically holds four PWR or nine BWR assemblies. Capacity would be dependent on the age and burnup of the specific fuel to be stored.

Figures IV-4 and IV-5 diagram storage of spent fuel in concrete storage casks designed for cooling by radiation and by natural convection.

In addition to the concrete cask storage system, the storage facility generally has the same types of auxiliary and support systems described for water pool storage except that (i) the fuel encapsulation facilities are more extensive inasmuch as it is necessary to encapsulate all fuel prior to storage and (ii) storage cask and canister maintenance facilities are required.

IV.B.1.4.2 Experience with Concrete Cask Storage

Long-term dry storage was investigated extensively in the United States as an alternative means for storage of solidified high-level waste, and the use of concrete casks stored at grade was favored as the system for exploitation. Such a storage system could be used to store spent fuel, and

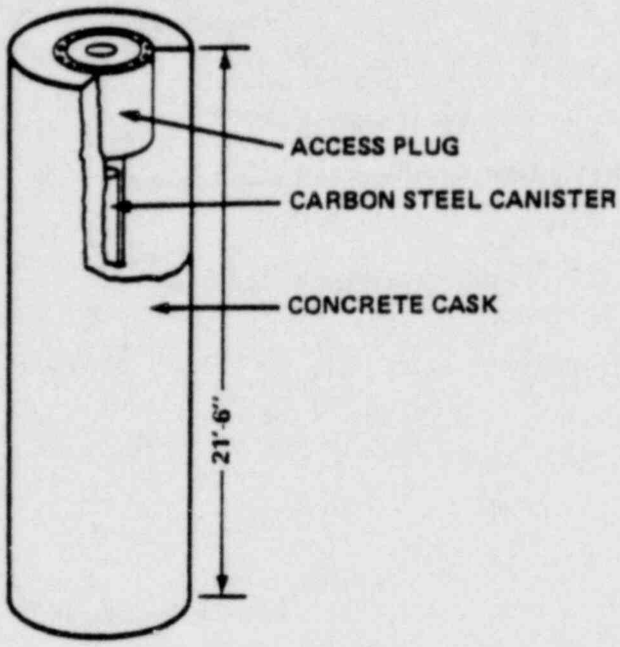


Figure IV-4. Storage of Spent Fuel in Sealed Concrete Storage Cask Cooled by Radiation

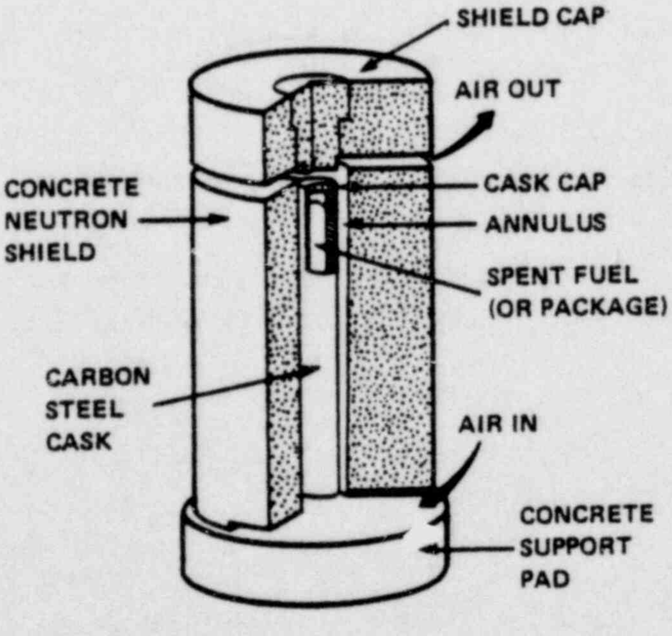


Figure IV-5. Storage of Spent Fuel in Concrete Storage Cask Cooled by Natural Convection indeed, it is essentially the same as the system currently under development by the Canadians. A demonstration program at Whiteshell Nuclear Research Establishment (27, 28) by Atomic Energy of Canada, Limited, is presently being undertaken to develop concrete casks for interim dry storage of spent fuel from the Canadian heavy water-moderated reactors (CANDU fuel), designed to contain fuel safely for periods of 50 to 100 years without replacement of the casks. Test canisters and casks have been constructed and are undergoing development testing.

As a result of the earlier U.S. work, the Department of Energy and its contractors have well-developed preliminary designs of systems for dry storage of encapsulated high-level waste in concrete storage casks (which function as radiation shields) at grade; an electrically heated test unit simulating a complete concrete storage cask system has been under test for several years (29, 30). The technology that has been developed could equally be used for temporary storage of encapsulated spent fuel in a similar configuration, and a substantial amount of design work has been done on facilities for receiving, handling, encapsulation, and surface storage of spent fuel assemblies (31, 32). Several types of designs have been evaluated, including both convection cooled casks as well as sealed, conduction/radiation cooled casks. As was described in IV.B.1.3.2, additional tests of dry storage modules are currently under way, employing commercial reactor fuel (24).

IV.B.2 Preferred Method for Storage--Water Pool Storage

IV.B.2.1 Summary Status of Storage Options

Water pool storage is the preferred method of spent-fuel storage for the reasons elaborated in the following paragraph. It is widely employed, and the ongoing research and development work described in IV.D.4 is being done to confirm present projections of long-term integrity of the fuel in storage (i.e., 50 years or more) and to evaluate the implications of higher burnup on the long-term integrity of stored fuel.

The dry storage options (particularly the storage in caissons beneath grade and surface storage in concrete casks) that are being developed as alternatives to water pool storage have certain potential advantages in that they require less complex facilities. In such systems, heat removal from the spent fuel would be effected by natural cooling through convection and/or radiation, whereas in water pool or dry vault storage the cooling medium would have to be purified on a continuing basis, and the system would include equipment items which would require frequent operational and maintenance attention.

Engineering studies of the dry storage concepts at Hanford (29-32), the work at EMAD on encapsulation and dry storage (24), and the results of continuing development work in Canada (27, 28) could provide the basis for qualifying one or both of these options as a second-generation AFR storage facility if required. In addition, experimental work being conducted at NTS on storage of fuel assemblies in granite and at Hanford on storage in basalt will provide additional information to corroborate the heat transfer models developed for the caisson storage concept.

The research, development, and demonstration activities associated with the development of these techniques are primarily directed toward the development of more passive systems of storage which have low associated maintenance, with emphasis on refinements of the design of equipment for use in the system. Low maintenance during storage could be expected to result in lower costs for storage over long time periods, if it proves desirable to store spent fuel over longer periods.

An International Nuclear Fuel Cycle Evaluation (INFCE) working group considered the question of interim spent fuel storage, and concluded:

> Experience exists with wet storage of LWR and HWR spent fuel for periods up to 20 years for low burn-up fuel. No significant difficulties are expected in projecting spent fuel behavior in wet storage for longer storage times and higher burnups. Storage in waterfilled pools including the use of compact racks can be considered a proven technology. Dry storage techniques are being investigated as an alternative for extended interim storage (33).

IV.B.2.2 Basis of Preference for Water Pool Storage

There are several reasons for the preference for water pool storage at this time. The fact that water pool storage has been used successfully on a large scale in the United States for over 30 years is persuasive evidence that there is an extensive technological base for the design and operation of this type of facility. This long experience has resulted in well-developed operating procedures and demonstrated performance capabilities. The ability to handle spent fuel safely in the water environment is evident because no incident involving spent-fuel handling has resulted in release of radioactivity from a storage facility, and only one of 11 incidents (worldwide) which had been reported to the International Atomic Energy Agency (IAEA) as of mid-1976 resulted in significant contamination in the facility itself. In this case, the contamination level was sufficiently high that it was necessary to wait several weeks before the fuel handling machine could be decontaminated. In all other cases, there were no measureable releases of radioactivity (34).

Use of water as the storage environment also has several technical advantages:

1. Water has a high cooling efficiency. Water pool storage of LWR fuel is required in every LWR facility because of its high efficiency for the cooling of spent fuel freshly discharged from the reactor when the heat removal requirements of spent-fuel storage are at their maximum.

2. Water is a good radiation shield. Use of water as the shielding medium is convenient; since water is transparent, safe handling of the radioactive fuel is greatly simplified, and the opportunities for inadvertent overexposure of workers are reduced.

3. Spent fuel and water are compatible under much more rigorous temperature conditions than those which prevail during storage. Spent fuel is removed from a reactor after 3-5 years exposure to the reactor environment. Water is used as the coolant in the reactor. During the time fuel is in the reactor, both the internal temperature of the fuel and the temperature of the surface in contact with the cooling water are substantially higher than is the case with fuel stored in water pools (see Table IV-8, in IV.D.4). Thus it can reasonably be expected that spent fuel stored in water would be essentially unaffected under the conditions prevailing in the storage pool, especially after the fuel has cooled by radioactive decay.

In short, although any of the dry storage techniques described i, a technically feasible alternative for AFR storage, water pool storage is usable over the entire interim storage period and should offer the lowest number of licensing problems, due to the abundance of supporting technical information which is available, and the prior licensing experience. The procedural mechanisms for licensing other than water pool facilities are less well developed, due to the smaller amount of background experience which is available.

IV.C PERFORMANCE REQUIREMENTS FOR WATER POOL STORAGE

As a result of substantial background of experience with spent-fuel storage, performance requirements for storage facilities have been well defined, and regulations specifically relating to licensing of independent storage facilities are in the final stages of formalization. In addition, the Commission has issued regulatory guides to support the regulations, and industrial standards have been developed to provide specific guidance to designers, builders, and operators of water pool storage facilities. Thus, the Department and/or industry, as applicants, and the Commission are well prepared to engage in licensing processes for water pool facilities.

IV.C.1 Design Requirements

The basic design criteria for facilities for the storage of spent fuel, whether in water pools or by any of the other techniques described in IV.B.1, include these:

1. Barriers to the release of radioactive materials.
2. Shielding for radiation emanating from spent fuel.
3. Cooling to maintain the spent fuel at a sufficiently low temperature to prevent degradation of the fuel cladding thereby minimizing release of volatile fission products.
4. Accident prevention and accident mitigating systems.

5. Criticality prevention systems.

6. Systems necessary for efficient operation of the storage facility.

IV.C.2 Regulations, Standards, and Guides

The standards and codes that are applicable to spent fuel storage facilities employing water pool storage include both government regulations and industry standards. Not only are Federal regulations and guides in place to cover continued storage of spent fuel, but also industry-generated standards to amplify these government documents are nearly ready for use. These are described in the following paragraphs.

IV.C.2.1 Regulations and Regulatory Guides

Historically, spent fuel storage installations have been licensed as integral parts of either spent fuel reprocessing plants or nuclear power plants. Such plants have for a number of years been licensed under the applicable sections of 10 CFR Parts 30 (35), 40 (36), and 70 (37) and under 10 CFR Part 50 (38). Part 50 is the basic regulation governing the design, construction, and operation of complex nuclear facilities such as nuclear power reactors and irradiated fuel reprocessing plants; Part 70 governs the possession and use of special nuclear material, while Parts 30 and 40 cover byproduct (i.e., fission products and other radioactive materials resulting from the fission process) and source material (i.e., natural uranium and thorium).

None of these regulations is completely applicable to licensing of spent fuel storage facilities when they are separate from either reprocessing plants or nuclear power plants, for several reasons. The prospective duration of storage activity at an AFR is comparable to that at power reactors, but the requirements of 10 CFR 50 are unnecessarily stringent for an independent facility storing aged fuel. On the other hand, the quantity of material likely to be stored at an AFR is sufficiently high that the inventories of radioactive material and special nuclear material far exceed those in facilities normally licensed under Parts 30 and 70, making a somewhat more rigorous licensing procedure advisable.

The Commission therefore undertook the development of specific regulations and regulatory guides for AFR's when it became apparent that they were likely to be required. In October 1978, the Commission issued a Notice of Proposed Rulemaking, proposing a new regulation to be identified as 10 CFR 72 (4), which would specify procedures and requirements for issuance of licenses to store spent fuel in independent spent-fuel storage installations. The deadline for receipt of comments on the proposed regulation was set for 4 January 1979. This proposed regulation is specific to AFR storage facilities and is therefore less rigorous than 10 CFR 50, which applies to all activities at reactors and reprocessing plants, including the storage of spent fuel at such facilities.

In December 1978, the Commission issued for public comment Draft Regulatory Guide 3.44 to support the proposed 10 CFR 72, and to set forth the format and content of the safety analysis report to accompany an application for a license to construct and operate an independent spent fuel storage facility (39). As was the case with the proposed regulation, the guide was restricted in its applicability and covered only AFR storage.

These draft regulations and regulatory guides are in reasonably advanced form, because the Commission and its predecessor agency have been involved in consideration and promulgation of such regulations since 1974--the year in which the first independent spent fuel storage facility was proposed for consideration. In November 1974, the Atomic Energy Commission issued a first draft of a regulatory guide for the license application, siting, design, and plant protection for independent spent fuel storage installations (40). This was subsequently issued as a formal guide (41). Specific guidance on the design of spent-fuel storage facilities associated with reactors is set forth in Commission Regulatory Guide 1.13 (42), which provides specific requirements to be met by at-reactor storage facilities. Because these facilities are built to general structural design criteria mandated by 10 CFR 50, this regulatory guide would not provide sufficient information for the design of an AFR.

The Environmental Protection Agency has enacted regulations, contained in 40 CFR 190 (43), establishing maximum allowable contributions to environmental radioactivity from fuel cycle facilities, including spent fuel storage facilities. These were published in January 1977 and will become fully effective 1 January 1983.

IV.C.2.2 American National Standards Institute Standards

In addition to these Federal regulations and guides, a proposed American National Standards Institute (ANSI) standard is at present out for ballot action. This standard, titled "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)" (44), was drawn up by a standards committee of the American Nuclear Society (ANS-57.7) comprising a group of individuals with expertise in the design, construction, and operation of spent fuel storage pools. This standard provides detailed guidance to the designer of a spent fuel storage facility in terms of general design guidelines appropriate to the hazard presented by the spent fuel, together with specific design parameters intended to assist in both design and licensing efforts. It contains requirements and recommendations for the design of major structures, heating, ventilation, and air conditioning systems, and for equipment and processes involved in handling and storing spent fuel and the associated radioactive waste control and monitoring activities.

Another proposed ANSI standard relating to the siting of spent fuel storage pools has been prepared by a similar committee of experts and awaits ballot action (ANS-2.19) (45). This standard provides detailed criteria and guidance for the selection and evaluation of sites for location of spent-fuel storage facilities. The groups involved in writing these two standards worked in close communication to ensure that the criteria would be consistent.

IV.D ACHIEVING SAFE AND ENVIRONMENTALLY ACCEPTABLE WATER POOL STORAGE

This Chapter summarizes the licensing experience on water pool facilities, describes in some detail the methods by which the performance requirements set forth in IV.A.1 and the design requirements of IV.C.1 are met, summarizes the findings of environmental impact assessments, and, finally, reviews the experimental evidence relating to spent-fuel integrity during extended storage periods.

Based on the information presented here, it is concluded that water pool spent-fuel storage facilities, whether at-reactor or elsewhere, can be operated in a safe and environmentally acceptable manner for the projected time periods required prior to the availability of geologic disposal facilities. Water pool storage facilities for spent fuel have been licensed for operation by the Nuclear Regulatory Commission or Atomic Energy Commission for more than 20 years. Operated in conjunction with reactors or fuel reprocessing plants, these water storage pools have provided a great volume of information and data pertinent to spent-fuel storage and the safety thereof. In addition to the licensed commercial facilities, government-operated production reactors and fuel reprocessing plants have also employed water pool storage for spent fuel since the early 1940's.

The environmental impacts of such storage operations have been shown to be negligible (46, 47). The results of spent-fuel monitoring and examination programs have shown that spent fuel can be stored underwater for long periods of time without significant degradation of the fuel cladding. In the unlikely event that fuel cladding should degrade during such storage, the fuel can be encapsulated as necessary to provide for the continued safe and environmentally acceptable storage thereof indefinitely.

IV.D.1 Past Licensing Experience

A number of water pool storage facilities for spent nuclear fuel have been licensed by the Commission and its predecessor agency in the past, and such licensing is continuing at the present time. Draft regulations and regulatory guides have been developed specifically for AFR storage facilities and have been issued for comment; final regulations and guides are being prepared. The following items summarize licensing actions taken in the past in connection with water pool storage of spent fuel:

1. The Atomic Energy Commission, predecessor regulatory agency to the Nuclear Regulatory Commission, licensed the spent-fuel storage activities of the Nuclear Fuel Services, Inc. (NFS) West Valley, New York, reprocessing plant in 1966 and such license remains in effect (48). The NFS spent fuel storage pool has a current storage capacity of about 260 MTU (50) and is currently storing 163 MTU.

2. The Atomic Energy Commission also licensed the spent fuel storage activities of the General Electric Co. (GE) Morris, Illinois, reprocessing plant in 1974, and Nuclear Regulatory Commission licensed an expansion thereto in 1976 (50). The GE spent-fuel storage pool has a current storage capacity of about 700 MTU (49) and is currently storing 350 MTU. The 10 CFR 70 license is currently in the "timely renewal" stage.

3. Allied General Nuclear Services (AGNS) applied for a license for its reprocessing plant at Barnwell, South Carolina, including the associated spent-fuel storage pool in 1969. A construction permit was issued by the Atomic Energy Commission for this facility in 1970 as a result of an Atomic Energy Commission review of the safety aspects thereof. Subsequently, application was made by AGNS for a Part 70 license to operate the fuel storage pool and to receive spent fuel in advance of receiving authorization to operate the reprocessing plant. Processing of this application had progressed to the public hearing stage, when it was interrupted by Nuclear Regulatory Commission following termination of the Part 50 (facility) license as a result of the decision to suspend the GESMO* proceeding (51). No action appears to have been taken on the Part 70 license application since that time.

4. Although no operating license has yet been issued for the AGNS facility, no substantive unresolved licensing issues arose during the proceedings related to licensing of the spent fuel storage pool. The AGNS spent fuel storage pool has a current capacity of about 400 MTU (49).

---

*Generic Environmental Statement Mixed Oxide Fuel.

5. The first nuclear power reactor and its associated spent-fuel storage pool were licensed by the Atomic Energy Commission in 1959 (52). Since that time, 76 nuclear power reactors with their associated spent-fuel storage pools have been licensed for operation, 70 of which still hold such licenses (53). Moreover, commencing in 1974, nuclear power reactor licensees have applied for licenses to expand the capacity of existing spent-fuel storage pools by storage rack modifications and additions. Since that time, 42 license amendments for such expansions have been authorized by the Nuclear Regulatory Commission, as Table IV-5 shows. In addition, 12 applications for expansion were still pending as of 31 December 1979.

Table IV-5. Number of At-Reactor Storage Pool Capacity Expansions

Year	Number of License Amendments for Expansion of Capacity of Spent Fuel Storage Pools
1975	4
1976	9
1977	9
1978	15
1979	5
Total	42

From the foregoing, it can be seen that Nuclear Regulatory Commission/Atomic Energy Commission licensing of water pool storage of spent nuclear fuel has been routinely practiced since 1959 and reasonably can be expected to continue in the future.

IV.D.2 Meeting Safety Requirements

The discussion in IV.D.1 indicates the existence of a substantial background of engineering and design information on water pool storage. Each reactor license application includes the design information and prospective operating criteria for the spent-fuel pool in order to demonstrate that the storage operation will meet regulatory and safety requirements. Similarly, all three of the commercial reprocessing plants in the United States (NFS, GE, and AGNS, described in IV.D.1) submitted detailed design information to the Commission (or its predecessor agency) demonstrating the design methods to be employed in order to meet the applicable safety requirements (5-7), including those for the spent fuel pools. NFS and GE have now stored spent fuel for the past 13 years and 5 years, respectively, in a manner which has met the safety requirements of such activities, and are continuing to so store spent fuel at the present time.

Moreover, E.I. du Pont de Nemours prepared for the Department in late 1978 a conceptual design of a large AFR spent-fuel storage facility (8, 9), which had a storage capacity of 5,000 MTU. This conceptual design contains a preliminary safety analysis of the design to establish its capability for meeting the necessary safety requirements. The ability to meet the safety requirements set forth in IV.C has been demonstrated in practice for AFR water pool storage in facilities having capacities up to 700 tons, and the evaluation of the conceptual large pool showed no capacity-related factors which would compromise the safety of the large facilities. Twenty years of experience in operation of spent-fuel storage pools at commercial nuclear power reactors provides no evidence to question the validity of these conclusions in respect to the general safety of water pool storage of spent fuel. The following paragraphs describe typical methods that are employed in the design of spent fuel storage facilities to meet the safety requirements of existing and prospective regulations.

IV.D.2.1 Barriers to Release of Radioactive Materials and Radiation

Important elements of the design of facilities for the handling of radioactive materials are the provision of multiple barriers to the release of radioactive materials to the environment and the provision of sufficient shielding in the facilities to assure that radiation levels in areas accessible to plant personnel and the public are within acceptable limits. Design criteria require that releases of radioactive materials and radiation levels be as low as reasonably achievable (ALARA, 10 CFR 20.1(c)) (2).

IV.D.2.1.1 Multiple Containment Systems

Multiple containment systems are a feature of the design of spent-fuel storage facilities. These include the following barriers to the release of radioactive materials to the environment:

1. The fuel pellet constitutes the first barrier to release. Inasmuch as it is a high-fired ceramic oxide which is insoluble in water, it provides a substantial impediment to the dispersion of fission products.

2. The fuel cladding constitutes the second barrier to release. Any defects in fuel cladding would diminish the efficiency of this barrier; however, release of some radioactive material through small defects may occur and is effectively handled by the water purification system of the facility, and fuel with major defects can be encapsulated to provide the necessary barrier.

3. The storage pool and water purification system constitute the third barrier to release.

4. The building structure and associated ventilation system constitute the fourth barrier to release, although these systems are not generally intended to function as a high efficiency barrier as are the preceding ones listed.

IV.D.2.1.2 Radiation Protection

Shielding and other design features of the various systems involved in spent-fuel storage facilities that minimize personnel radiation exposure levels are as described below.

IV.D.2.1.2.1 Shipping Cask Handling, Cask Vent, Cooling and Flush System

Shipping casks used for the transportation of spent fuel are heavily shielded structures designed to meet the regulatory requirements relating to surface radiation levels. Thus, radiation levels near the cask during the various handling operations will be relatively low and will not require special personnel shielding considerations in the design of the facility. Provisions will have to be made to prevent exposure to the hot cask surfaces after removal of impact structures, heat shields, and other hardware.

Casks vent to an off-gas system that contains a condenser, liquid collection tank, carbon absorption bed, high-efficiency particulate air filter, and vent gas sampling system that permits radioactivity tests to detect unexpected fuel failures.

Removal of surface dirt on the casks is accomplished by washdown, the residue of which drains to the low-level waste treatment facilities.

If the fuel requires cooldown prior to immersion in the fuel unloading pool, the cooling is accomplished by injection of steam followed by water to reduce the temperature of the fuel gradually. The cooling water is collected in a hold tank for subsequent disposal in the liquid waste system. Flushing the cask interior with water may be necessary to reduce the level of contamination prior to unloading in the fuel unloading pool. The vent, cooling and flush piping system, and tank will have shielding that reduces radiation levels to below the required limits.

IV.D.2.1.2.2 Fuel Unloading Pools

Unloading of spent fuel from the shipping cask is accomplished in the fuel unloading pool, where the water level over the fuel provides the shielding necessary to reduce surface radiation levels to the levels required by regulation. Safety measures in the form of positive controls over water level in the pool and mechanical devices to prevent hoisting of the fuel above a safe level are provided. These measures ensure that there is always adequate water shielding over the fuel. Another design safety measure is the avoidance of any piping so arranged that it might permit siphoning of water from the pool.

To prevent leakage of water from the pool, a stainless steel liner is provided, with leak channels behind all weld points to collect any leakage and transfer it back to the pool water system; monitoring of these leak channels provides warning of leakage. Breaching of the liner by the unlikely event of a dropped cask is prevented by an impact limiting structure in the bottom of the pool and means of preventing cask tipping.

Failed fuel and "crud" on the surface of the spent fuel may contribute radioactive material to the pool water. To control the radioactivity of the pool water, a filter-deionizing system is provided to remove particulates and dissolved radioactive contaminants and to maintain the pool water activity at low levels.

IV.D.2.1.2.3 Fuel Storage Pools

All the special design considerations described above for the fuel unloading pools also apply to the fuel storage pools, except that the impact limiting structure is not required. In addition, the water recirculation system includes a heat exchanger to maintain the fuel storage pool water temperature below 40°C when the storage pools are filled with spent fuel to the design capacity. The fuel pool volume provides a heat sink, which provides time to reestablish cooling operations in the event of a temporary loss of the external cooling system. Backup water supplies such as on-site wells, ponds, or lakes provide makeup water from evaporative loss in the event of boiling. The emergency water supply system must be designed to meet seismic and tornado design criteria.

IV.D.2.1.2.4 Water Purification System

The filter-deionizer and deionizer waste disposal systems for the fuel unloading and fuel storage pools will normally contain large amounts of radioactivity requiring shielding. These facilities are located in concrete shielded cells and are operated by remote control. Contact (hands-on) maintenance for some of these facilities is possible after flushing and decontamination to reduce radiation levels. Equipment requiring extensive decontamination before contact maintenance is replaced (to reduce downtime) and transferred to a shielded cell for decontamination.

IV.D.2.1.2.5 Waste Handling Systems

The liquid waste disposal system includes feed tanks, pumps, and evaporators for collection and concentration of the various radioactive liquid wastes generated in the operation of the facility. These facilities require moderate concrete shielding. Manual valves are operated by extension handles through sleeves in the shield wall where feasible; elsewhere, remotely operated valves are used. Contact maintenance is possible after emptying and cleaning the equipment to reduce the radiation levels to below the required limits.

IV.D.2.2 Criticality Prevention

All spent fuel handling, transfer, and storage operations are designed to limit the fuel to a subcritical configuration. The number of assemblies being transferred and the storage array in the pool is limited to that which restricts reactivity to $K_{eff} \geq 0.95$ at the 95% confidence level based on unirradiated fuel specifications.

Proposed regulation 10 CFR 72 would require use of a "double contingency principle" to ensure subcriticality at all times during fuel handling and storage activities. This principle requires that no single failure of a protection system can result in criticality. Conversely, two (or more) independent failures must occur before a criticality can occur. Protective systems may be either positive mechanical design features which preclude the assembly of a critical system, or administrative methods which require specified procedural actions prior to or while undertaking an activity involving handling or transfer of materials. For a water pool storage facility, these are illustrated by the following:

1. Design features: (i) Geometry control of the storage array, i.e., assembly-to-assembly distance controlled by the structure of the storage racks; (ii) provision of poison curtains between adjacent fuel assemblies; (iii) mechanical limits on cranes which prevent the transport of heavy objects (e.g., casks) over stored fuel; (iv) mechanical devices on fuel transfer racks which

preclude dropping of the rack or spilling its contents onto the storage array in the event of any single failure of the hoisting system during transfer of fuel; and (v) control instrumentation on critical processes.

2. Administrative features: (i) Procedures requiring verification of the enrichment of fuel prior to placing in storage; (ii) procedures requiring periodic checking of the continuing presence of poison in the storage racks; (iii) procedures for control of maintenance work on handling systems to ensure that no compromise of the effectivity of mechanical design safety features occurs; (iv) procedures for control of design changes to ensure that no change is made which compromises any safety feature; and (v) operating procedures which specify how each operation involving handling of spent fuel is to be conducted.

Wherever feasible, the primary criticality control must be achieved by means of positive mechanical design features; administrative features may provide backup protection.

In addition to the features identified above, fuel storage facilities are provided with nuclear criticality monitors and alarms in areas where there is potential for nuclear criticality.

IV.D.2.3 Accident Prevention and Mitigation

IV.D.2.3.1 Operating Errors

The design ensures that no single operating error can cause an incident having unacceptable safety consequences. For example, fuel handling equipment is designed to ensure that no single operating error causes a nuclear criticality event in the fuel storage arrays.

The authority to manipulate safety-related equipment and controls is limited to trained personnel or, in an emergency situation with direct supervision, to an individual with adequate training in such operation. Supervisory personnel who direct the manipulation of safety-related equipment and controls are required to have a level of training in such operations comparable to that of trained operating personnel.

IV.D.2.3.2 Seismic and Tornado

Proposed 10 CFR 72 would establish a requirement that the design basis earthquake be assumed to have a peak horizontal ground acceleration of not greater than 0.25 g with a recurrence interval of at least 500 years. This recurrence interval is equivalent to a 90% probability of not being exceeded in 50 years. An AFR storage site would be selected such that the site would meet the aforementioned 0.25-g limit; if a site of interest showed a greater peak acceleration, it would be subject to a site-specific analysis in accordance with 10 CFR 100. Equipment and facilities that are designed to withstand the design basis earthquake are those described in Table IV-6.

The design basis tornado is assumed to have a maximum wind speed of 360 mph. Equipment and facilities that are designed to withstand the design basis tornado are those described in Table IV-7.

IV.D.2.3.3 Fire

Facilities are designed to minimize fire potential. No significant quantities of highly flammable or explosive chemicals or materials are used at such facilities. The design includes a fire detection system, alarms, and fire control equipment.

IV.D.2.3.4 Vent Stack

Vent stacks for nuclear facilities are so designed and located with respect to critical buildings that the collapse or fall of a stack, due to natural phenomena or other cause, would not result in damage to any critical facility.

Table IV-6. Facility Design Criteria for Design Basis Earthquake

Equipment	Performance Criteria
Fuel unloading and storage pools	No loss of functional integrity
Storage basket anchors	No loss of functional integrity
Cranes	Cranes would not collapse into storage pools nor drop heavy equipment onto stored fuel.
Building structure	Building structural steel would not collapse onto pools.
Emergency makeup water source and delivery system	No loss of functional integrity; delivery system would be accessible for use following an earthquake.
Pool level monitoring instrumentation	No loss of functional integrity

Table IV-7. Facility Design Criteria for Design Basis Tornado

Equipment	Performance Criteria
Fuel unloading and storage pools	No loss of functional integrity
Storage baskets anchors	No loss of functional integrity; damage of fuel from tornado missiles would not cause radioactive release above Federal guidelines.
Cranes	Cranes would not collapse into storage pool nor drop heavy equipment onto stored fuel.
Building structure	Heavy building structural steel would not fall or be blown onto pools.
Emergency makeup water source	No loss of functional integrity
Deionizers and filters	No dispersal of contaminated resin
Evaporators	No dispersal of evaporator bottoms
Solid waste	Drums caged to prevent dispersal of drummed waste.

IV.D.2.3.5 Emergency Power (Equipment and Lighting)

The design provides for emergency power in the form of diesel generators and/or batteries for lighting, communication systems and safety-related monitors, alarms, and equipment. In addition, emergency power will supply that process equipment deemed essential to safe shutdown of the facility on loss of normal power.

IV.D.2.4 Ventilation

IV.D.2.4.1 Building Ventilation System

The ventilation system designs comply with standard industrial practice to provide for protection, comfort, and safety of operating personnel during normal operation. The systems restrict the spread of radioactive contamination within the facilities, and any release of radioactive particulates to the environs will not exceed that allowed in the Federal guidelines.

The process areas ventilation system provides once-through flow of air. Inlet air passes through roughing filters to remove the bulk of particulate matter in ambient air. Air locks at entrances to ventilation zones provide control of the air flow balance between zones. Flow of air is from zones with lesser potential for radioactive contamination to areas with higher potential for contamination.

With a few exceptions, ventilation air from all areas of a building exhausts unfiltered to the atmosphere via a stack. The exceptions are the shielded process areas, solid waste handling area, and regulated shops. Exhaust air from these areas pass through a single-stage, high efficiency particulate air (HEPA) filter before discharge to the atmosphere via the stack.

IV.D.2.4.2 Process Off-Gas System

The cask vent s;..em has a carbon absorption bed for removal of volatile fission gases and a single stage of HEPA filters before discharging the off-gas to the stack. Failed fuel containers are available to collect and contain fission products released from damaged fuel assemblies in the unlikely event that they would be required. Any fission gases from this collection system are routed to the cask vent off-gas system.

IV.D.2.5 Detection and Monitoring Systems

Designs normally include the following radiation detection and monitoring systems.

IV.D.2.5.1 Radiation Monitors

Radiation measuring and recording monitors are located throughout the pool building.

IV.D.2.5.2 Air Monitors

Facilities are provided for sampling the ventilation air in the building and for monitoring airborne radioactive contaminants. Stack monitors measure and record radioactive releases to the atmosphere.

IV.D.2.5.3 Pool Level Leak Detection System

Pool water level and leak detection collection systems provide a check on integrity of the stainless steel liners in the fuel unloading and fuel storage pools.

IV.D.2.5.4 Nuclear Incident Monitoring System

A nuclear incident monitoring system provides for detection of a criticality incident. The system uses redundant monitors at each potential criticality location. American National Standards Institute Standard N16.2-1969 (American Nuclear Society Standards Committee ANS-8.3) provides additional design guidance.

IV.D.2.5.5 Environmental Monitoring Systems

A system of monitoring stations and wells permits sampling and monitoring of vegetation, soil, and surface and ground waters for radioactivity.

IV.D.3 Environmental Considerations

Both the Commission and the Department have assessed the environmental impact of storing spent nuclear fuel. The Commission considered the possible shortage of spent fuel storage capacity and the options for dealing with the problem from the standpoint of long-term regulatory policies. The NRC final EIS on the storage of spent fuel was issued in August 1979 (1). This statement concluded, "The storage of LWR spent fuel in water pools has an insignificant impact on the environment, whether such pools are at reactor sites or away therefrom." It also concluded, "The technology of water pool storage is well developed . . . radioactive waste that is generated is readily confined and presents little potential hazard to the health and safety of the public (1)."

The Department's environmental analysis addressed the impacts of implementing or not implementing the United States' announced policy of providing interim storage for nuclear power reactor spent fuel, of both domestic and foreign origin. In August 1978, the Department issued for comment the Draft EIS on the storage of domestically generated power reactor fuel (47), which was subsequently supplemented in December 1978 (54). A final EIS is in preparation. The Draft EIS notes:

In summary, the environmental impacts from all alternatives considered, either from implementing or not implementing the spent fuel storage policy, are nominal: The decreased resource consumptions and environmental impacts of alternatives that assume reactor basin operation at less than full-core reserve must be balanced against the reduced flexibility in reactor operation and the possibility of forced shutdowns which could lead to the use of higher-cost supplemental power or reduction of electrical power generation. At-reactor storage increases environmental effects compared with ISFS basin storage because additional storage basins are constructed and operated. However, the impacts are relatively small compared with available resources and risks from natural radiation sources (54).

IV.D.4 Assessment of Extended Storage Performance

The purpose of this section is to show that fuel can be stored in pools as received from reactors over the expected time span until geologic disposal is available. Much of the spent-fuel experience base has developed at reactor pools. Experience at BWR pools is directly relevant to AFR pools because both types have deionized water chemistries. PWR pools use dilute boric acid chemistries to provide compatibility with the boric acid chemistry of the PWR primary system coolant during refueling.

Nuclear fuel survives extended reactor exposures (normally 3 years) to water under intense radiation and elevated temperatures before discharge to the storage pool. By comparison, the fuel assembly materials are exposed to much more benign conditions in water storage (Table IV-8). Water pool storage involves relatively simple technology. Because water storage provides a key interim role in nuclear fuel management, it is important to state the basis for the judgment that it is a viable technology for storage of water reactor fuel. A major element of that conclusion arises from an assessment of current condition of the spent-fuel inventory. To date, there is no evidence that commercial water reactor cladding has deteriorated. Zircaloy-clad oxide fuel has survived up to 20 years in water storage. Stainless-clad water reactor fuel has survived up to 12 years in water. The second element of assurance involves an intention to continue surveillance of spent fuel for as long as pool storage is required. The final element of assurance is the recognition that even if severe fuel assembly deterioration were to develop, isolation of assemblies by encapsulation (canning) is a demonstrated alternative.

IV.D.4.1 Characteristics of Typical Spent Fuel

General characteristics of the spent fuel of United States origin are discussed in II.C.3 of this report. For reference here, the configurations of fuel rods, BWR assemblies, and PWR assemblies are presented in Figures IV-6 through IV-8. Characteristics of typical LWR (BWR and PWR) fuel assemblies are described in Table IV-9.

Some discussion of Canadian pressurized heavy water reactor (PHWR) fuel storage experience is included. PHWR fuel rods are Zircaloy-clad uranium oxide (55). Therefore their storage behavior has substantial relevance to United States LWR fuel, even though the geometry differs from LWR fuel geometries (see Table IV-9 for LWR fuel assembly dimensions). The PHWR fuel assemblies have the following dimensions (56): ~50 cm (22 in) long; 8 to 10 cm (3.2 to 4.0 in) in diameter; contain 19 to 37 fuel rods, each with a diameter of ~1.5 cm (0.59 in). The PHWR assemblies weigh 17 to 25 kg (37 to 55 lb). The maximum burnups at discharge are about 10,000 MWd/MTU, which is below burnups for the majority of United States fuel. However, the Canadian experience is a valuable source of spent-fuel storage information for the reasons listed below:

1. The cladding is Zircaloy.

2. The fuel pellets are UO_2.

3. Canadian fuel burnups overlap with lower burnups on United States fuel.

4. Canadian spent fuel has been examined after up to 16 year of pool residence (see IV.D.4.3). Some PHWR fuel is st functioning satisfactorily in the NPD reactor after years at reactor primary system conditions.

Table IV-8. Comparison of Conditions for Light Water Reactor Fuel

Condition	In-Reactor	In-Pool
Fuel temperature, °C (peak centerline)	1,200-1,350	100
Water temperatures		
BWR, °C	270-300	20-50
PWR, °C	320-340	20-50
Cladding (inside) surface[a]	340-400	30-60
Gas pressure[b]		
BWR, bar	4.8-48.0	2.0-20.0
PWR, bar	39.0-150	17.0-85.0
Calculated fission gas evolution[c]		
BWR	2%	negligible
PWR	15%	negligible
Surface heat fluxes, W/cm^2	Up to 80	0.03[d]
Radiation fluxes (maximum)[e]		
neutron, $n/cm^2/s$, >1MeV	$3\text{-}6 \times 10^{13}$	10^5
gamma, R/hr	10^9	10^6

[a]After cooling for several weeks, the exterior surface temperature is 10°C above the bulk water temperature; the interior and exterior cladding surface temperatures are essentially the same at pool storage conditions.

[b]For intact fuel rods.

[c]Based on modeling calculations, measured values may be lower.

[d]After cooling for 1 yr.

[e]Upon discharge; decreases exponentially.

Source: (Reference 57) A.B. Johnson, Jr., "Spent Fuel Storage Experience," Nucl. Tech., Vol. 43, pp. 165-173, 1979

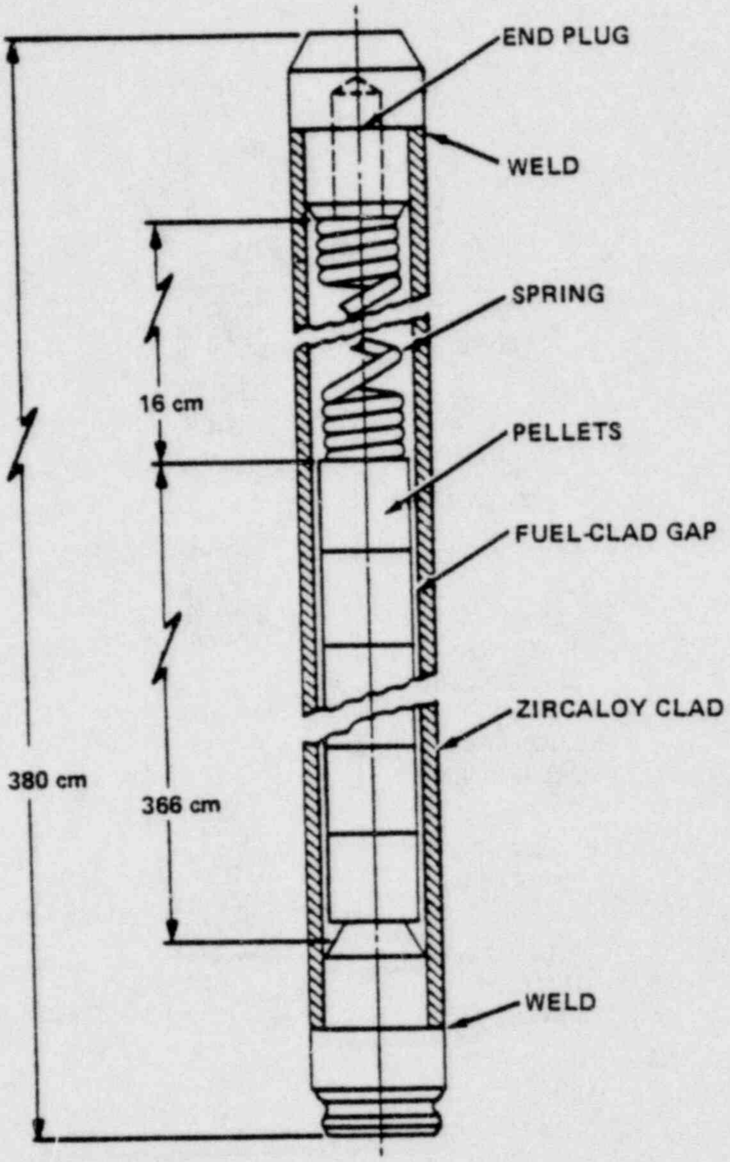


Figure IV-6. Typical Fuel Rod

Source: (Reference 58) W.H. Baker and F.D. King, Technical Data Summary, Spent-Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant, DPSTD-AFCT-77-7, Savannah River Laboratory, p. A.3, Aiken, SC, August 1977

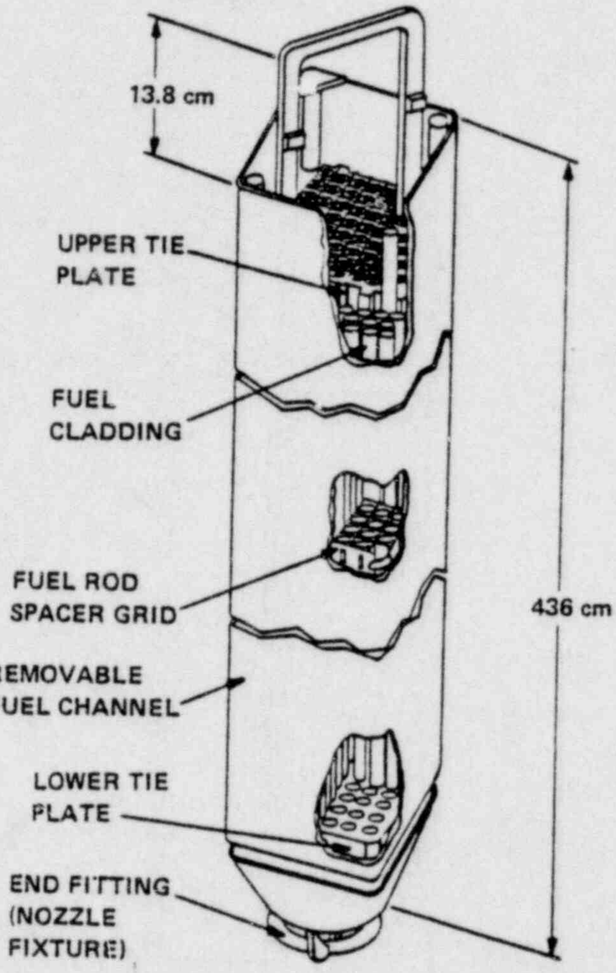


Figure IV-7. Boiling Water Reactor (BWR) Fuel Assembly

Source: (Reference 58) W.H. Baker and F.D. King, Technical Data Summary, Spent-Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant, DPSTD-AFCT-77-7, Savannah River Laboratory, p. A.4, Aiken, SC, August 1977

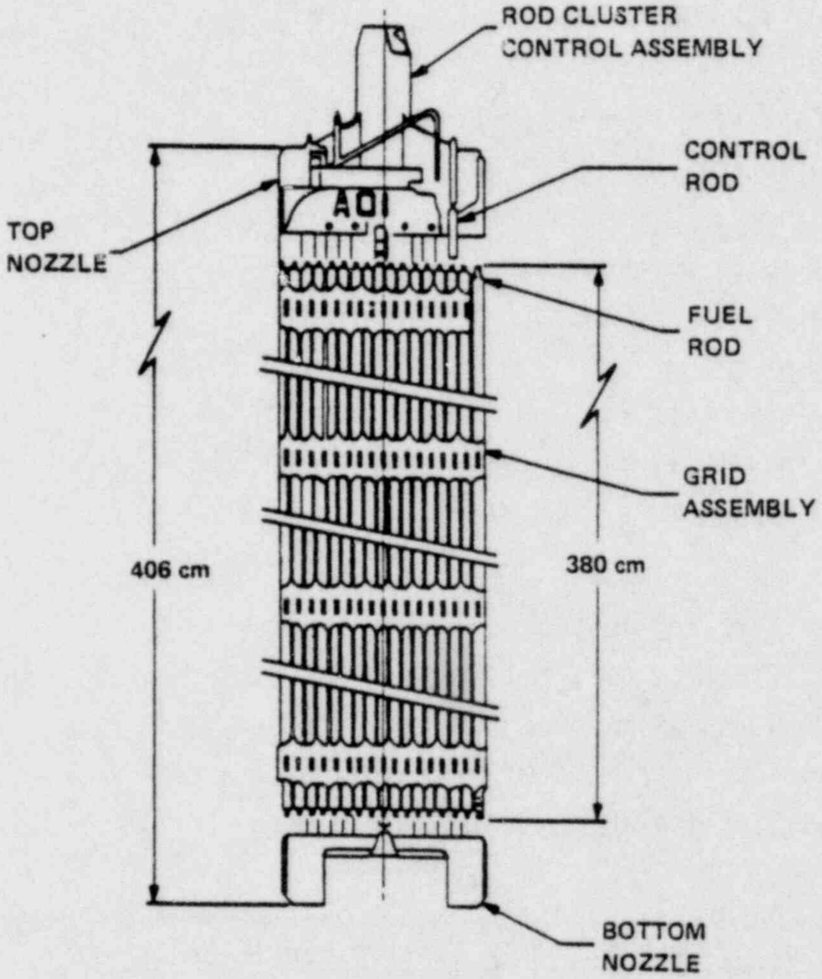


Figure IV-8. Pressurized Water Reactor (PWR) Fuel Assembly

Source: (Reference 58) W.H. Baker and F.D. King, Technical Data Summary, Spent-Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant, DPSTD-AFCT-77-7, Savannah River Laboratory, p. A.6, Aiken, SC, August 1977

Table IV-9. Characteristics of Typical Light Water Reactor Fuel Assemblies

Characteristic	PWR	BWR
Overall assembly length, m	4.1	4.5
Cross section, cm	21.4 x 21.4	13.9 x 13.9
Fuel rod length, m	3.8	4.1
Active fuel height, m	3.7	3.8
Fuel rod OD, cm	0.95	1.2
Fuel rod array	17 x 17	8 x 8
Assembly total weight, kg	670	279
U/assembly, kg	460	190
UO_2/assembly, kg	525	215
Zircaloy/assembly, kg	130[a]	57[b]
Hardware/assembly, kg	16[c]	8[d]
Total metal/assembly, kg	145	65

[a]Includes Zircaloy control-rod guide thimbles.

[b]Includes Zircaloy fuel-rod spacers.

[c]Includes 10-kg SS nozzles and 5.5-kg Inconel-718 grids. (Inconel = trademark of International Nickel Co.)

[d]Includes SS tie plates and negligible amount of Inconel springs.

Source: (Reference 59) U.S. Department of Energy, Analytical Methodology and Facility Description - Spent Fuel Policy, DOE/-ET0054, p. I-4, August 1978

IV.D.4.1.1 Assembly Components and Materials of Construction

The fuel assembly materials used in the United States are summarized in Table IV-10 for LWR fuel. Canadian fuel assemblies are constructed entirely of Zircaloy except for the beryllium braze used for wear pads. The fuel assembly materials have good corrosion resistance in water, both at reactor operating conditions and under pool storage conditions.

Table IV-10. Fuel Assembly Materials

Component	Subcomponents	Material	Alloy
Fuel rods	--	Zircaloy	Zry-2 (BWR) Zry-4 (BWR)
		Stainless steel	304 348H
Fuel spacers	Grid	Stainless steel	304
		Inconel	718
		Zircaloy	Zry-4
	Springs	Inconel	718
Upper tie plates	Bail/tie plate	Stainless steel	304
	Bolts/nuts	Stainless steel	304
		Inconel	600
	Springs	Inconel	718
Lower tie plates	Tie plate/nozzle	Stainless steel	304
Tie rods		Zircaloy Stainless steel	--

Source: (Reference 15) A.B. Johnson, Jr., Behavior of Spent Nuclear Fuel in Water Pool Storage, p. 47 BNWL-2256, Battelle Northwest Laboratories, Richland, WA, 1977

IV.D.4.1.2 Radionuclide Content

The radioactivity inventory in a fuel assembly has the following sources:

1. Fission Products Contained Within the Fuel Cladding--Power generation involves fissions of uranium and plutonium atoms (isotopes) producing smaller atoms which are predominantly radioactive; some have short lives (seconds, minutes, or hours) and decay to inconsequential concentrations almost before the assembly is discharged from the reactor. Other isotopes have half-lives in the time scale of days or years. Isotopes with significant activities over the expected period of pool storage have been calculated (60).

Most of the fission products remain in the fuel pellets, but some, principally krypton, reside as gases inside the fuel rod during pool storage.

2. Transuranics--A fraction of the uranium absorbs neutrons and transmutes to atoms with atomic numbers higher than uranium. Activities of major transuranics in the spent fuel have been calculated (61).

3. Activation Products--The fuel cladding and assembly hardware (end plates, fuel spacers, etc.) contain isotopes which become radioactive when irradiated by neutrons. The principal isotopes generated in the fuel assembly metals which are concern over the projected storage interval are ^{60}Co and ^{63}Ni (62).

4. Crud Layers--Corrosion products from the reactor coolant system (principally iron-base and nickel-base alloys) are transported to the reactor core, where they settle on the fuel assembly surfaces and form a "crud" layer, comprising radioactive oxides. The principal isotopes are the same as those in the activated iron-base and nickel-base fuel assembly metals (Table IV-13). However, some fuel crud deposits contain other isotopes such as ^{65}Zn. If exposed to a reactor where fuel failures have occurred, the crud layers will absorb fission products such as iodine, cesium, and strontium.

The radioactive nuclide inventories and thermal powers of spent fuel from a typical large PWR have been calculated and published (63). Composition of BWR spent fuel is generally similar, but the radioactive nuclide inventory and thermal power are generally less because of lower average burnup and specific power in comparison with typical PWR spent fuel. For conservatism, the Draft EIS on the spent-fuel policy (47) assesses the environmental effects of spent LWR fuel management by using fission product inventories and thermal powers (per MTU) calculated for PWR conditions.

IV.D.4.1.2.1 Radioactivity Versus Time

The radioactivity of an LWR spent fuel assembly declines exponentially as a function of time; in the period of interest for storage of spent fuel, the surface dose rate in rem/hr declines from approximately 2×10^5 rem/hr at 1 year out of the reactor to approximately 9×10^3 rem/hr 50 years after discharge (Table II-3).

IV.D.4.1.2.2 Heat Output Versus Time

During radioactive decay, alpha and beta particles and electromagnetic radiation emit from unstable nuclei. Absorption of the emitted species in the surrounding matter causes temperature increases. Heat generation rates depend on the nuclide inventory and decay characteristics. Typical heat generation rates for BWR and PWR fuel as a function of time appear in Table IV-11.

Table IV-11. Effect of Decay on Thermal Power of Spent Fuel[a]
(Operating power = 30 MW/MTU, exposure = 33,000 MWd/MTU)

Time After Discharge (years)	Approximate Thermal Power: Watts/MTU	Watts/Assembly: PWR	Watts/Assembly: BWR
0.5	16,700	7,700	3,200
1.0	10,200	4,700	1,900
2.0	5,400	2,500	1,000
4	2,500	1,150	470
6	1,700	760	320
8	1,300	610	250
10	1,200	550	230
30	720	330	140

[a]BWR fuel assumed to be irradiated under PWR conditions for conservatism in assessing environmental effects of spent fuel management.

Source: Adapted from Table II-4 in Chapter II.C.

IV.D.4.1.3 Burnup/Exposure

Burnup is a measure of the amount of fuel which was converted into energy. It also is proportional to the number of neutrons which interacted with the fuel cladding. There has been a continuous increase in the average burnup of fuel at discharge. The average was about 8,000 MWd/MTU* in 1962 (64). In January 1973, the average burnup of all United States discharged fuel was 11,200 MWd/ MTU, and the worldwide average burnup of discharged Zircaloy-clad fuel was 8,200 MWd/MTU (65). In January 1973, the highest discharge burnup for Zircaloy-clad fuel, on a world-wide basis, was 16,700 MWd/MTU for BWR's and 25,200 MWd/MTU for PWR's (66) for quantities of fuel of 5 MTU. The average discharge burnup of domestic fuel was 24,000 MWd/MTU in 1978 (64). The current nominal design burnups for domestic PWR fuel and BWR fuel are 33,000 MWd/MTU and 27,300 MWd/MTU, respectively (66, 67).

In the future, batch-average discharge burnups in the 40,000 to 55,000 MWd/MTU range may develop (67, 68). Consequently, future discharges of spent fuel will have higher values of thermal powers. The effect of these higher values will be taken into account in spent-fuel pool designs. With extended burnups, peak burnups in the fuel pellets would exceed 60,000 MWd/MTU (up from 40,000 MWd/MTU) (68, 69). To date, the highest burnups in commercial LWR fuel are 62,000 MWd/(rod average) MTU, on rods from demonstration fuel exposed in the Zorita reactor (70) (Spain). Some of the high-burnup rods were examined (71). Others are now in dry storage in Spain.

IV.D.4.1.4 Condition of Assembly After Reactor Exposures

Several characteristics of a fuel assembly change during the reactor irradiation:

1. Oxide film growth occurs: typical oxides on discharged Zircaloy-clad fuel are 15 to 25 μm of ZrO_2 on BWR fuel rods; 15 to 20 μm of ZrO_2 on PWR fuel rods; local oxide thicknesses range from 50 to 160 m.

*Megawatt-days of thermal energy released by fuel containing 1 metric ton (10^6gm) of heavy-metal atoms (MWd/MTU).

2. Zircaloy cladding and spacers absorb and retain substantial hydrogen; typical hydrogen concentrations in spent-fuel cladding are 60 to 100 ppm (PWR) and 60 to 120 ppm (BWR). Most of the tritium released from the fuel also is absorbed in and retained by the Zircaloy (stainless steel cladding retains very little hydrogen).

3. Species circulating in the reactor coolant deposit on the fuel assembly surfaces, forming a superficial layer of oxides (crud).

4. Metal strength increases and ductility decreases during neutron irradiation.

5. Fuel rod length increases slightly.

6. The uranium oxide fuel tends to swell during irradiation; cracks develop in some fuel pellets.

The considerations in post-irradiation fuel assembly condition pertinent to water storage are summarized below:

1. The oxide films which form on fuel assembly metals at reactor temperatures are stable at pool storage temperatures; they do not thicken perceptibly or spall during water storage.

2. Crud species occur in two forms; loose and tenacious; loose species are more common on BWR fuel (principally Fe_2O_3); some radioactive crud particles may detach from the fuel surfaces during fuel handling and storage; if fuel failures occurred during reactor residence, fission products absorbed on the crud layers will desorb slowly during pool storage.

3. Mechanical properties of spent fuel assembly materials are fully satisfactory to endure movements during fuel handling operations.

4. Stresses in the fuel cladding are considered to be low relative to those which were present at reactor operating conditions because the fuel tends to shrink away from the cladding upon cooling and because gas pressures inside the fuel rods decrease upon cooling.

IV.D.4.2 Spent-Fuel Storage Experience

A great amount of experience with water pool storage of spent fuel assemblies and other radioactive materials has been acquired since the early days of the Manhattan project (1943-44). This experience is pertinent to the assessment of the suitability of continued water pool storage of spent nuclear fuel. The operational experience with fuel for water-cooled reactors, including discharged fuel, in the western world is based on more than 9 million fuel rods (including 3.5 million short Canadian fuel rods) (72).

Swedish investigators participating in the Karnbanslesakerhet (KBS) study have proposed a plan for handling spent nuclear fuel which involves storage in the reactor pool for at least 6 months, followed by transfer to a central water pool storage site for 40 years storage. After this storage period, the fuel would be encapsulated and transferred to a repository for disposal. In considering the acceptability of long term storage of zirconium alloy-clad fuel, these investigators concluded:

1. It is technically feasible and permissible from the viewpoint of safety to shorten or lengthen the storage period, and

2. No degradation mechanisms which could affect the integrity of the fuel within a period of 40 years have been identified (73).

The early fuel was reprocessed, but some Zircaloy-clad spent fuel has remained in water storage for two decades. This section summarizes some of the longer exposures of spent-water reactor fuel in water pools and indicates characteristics of the stored fuel. It also summarizes storage experience for fuel which has reactor-induced defects. Finally, it indicates impacts of damage which occurred during underwater handling of spent fuel.

IV.D.4.2.1 Storage History of the Current Spent Fuel Inventory

An assessment of the storage characteristics of the current spent-fuel inventory is important to project probable future storage behavior. The current scope of U.S. and Canadian water reactor spent-fuel storage appears in Table IV-12. The oldest and highest burnup fuel is summarized in Table IV-13. To date there has been no evidence that the commercial water reactor fuel is degrading during water storage.

Table IV-12. Spent-Fuel Inventories in Pool Storage in the United States and Canada Through December 1979

Reactor Type	Spent Fuel Assemblies in Water Storage
BWR[a, c]	14,600
PWR[a, c]	8,150
PHWR[b]	129,000

[a]See Table IV-9 for assembly size.

[b]Estimated December 1979 inventory for Bruce and Pickering Stations; assemblies are ~50 cm long and weigh up to ~25 kg.

[c]Based on information compiled by S.M. Stoller Corp. The numbers include all U.S. commercial LWR fuel discharged. The numbers include approximately 1,135 BWR assemblies and 480 PWR assemblies which were reprocessed at the Nuclear Fuel Services Plant, West Valley, N.Y., 1967-1971. The BWR fuel discharges represent 2,540 MTU; the PWR fuel discharges represent 3,380 MTU. Section V.A.1 shows the inventory of stored fuel as a function of time.

Table IV-13. Characteristics of the Current Spent-Fuel Inventory

Characteristic	Cladding	Reactor/Type	MWd/MTU Burnup	Storage Period in Water	Remarks
Longest residence in-pool	Zircaloy-2	Shippingport/PWR	6,000	1959 to present[a]	Stored in DIW[b]
	Stainless steel	VBWR/BWR	10,000	1963-1975	Stored in DIW; reprocessed 1975
	Zircaloy-2	BWR	25,000	1974 to present	Stored in DIW
	Zircaloy-4	PWR	17,000	1973 to present	Stored only in H_3BO_3 [c] pool
	Zircaloy-4	PWR	7,700	1971 to present	Stored briefly in H_3BO_3 pool; then stored in DIW
	Stainless steel	PWR	37,500	1976 to present	Stored only in H_3BO_3 pool
	Stainless steel	BWR	22,000	1975 to present	Stored in DIW
Highest burnup	Zircaloy-4	Zorita/PWR	62,000	1975-1976	Rods in dry storage since 1976
	Zircaloy-4	KWO/PWR	39,000	1975 to present	Stored briefly in H_3BO_3; then stored in DIW
	Zircaloy-2	BWR	27,300	1977 to present	Stored only in DIW
	Zircaloy-2	PWR	41,000	1974 to present	Resided in reactor 1957 to 1974; then DIW pools
	Stainless steel	BWR	22,000	1975 to present	Stored only in DIW
	Stainless steel	PWR	37,500	1976 to present	Stored only in H_3BO_3 pool

[a]One Shippingport assembly stored since 1959; one stored since 1961; one stored since 1964.
[b]DIW: deionized water; applies to AFR and BWR pools; also to Shippingport PWR pool.
[c]Boric acid: applies to PWR pools; concentration - 0.18 to 0.21 moles/liter boric acid.

IV.D.4.2.2 Storage History of Fuel That Developed Defects In-Reactor

A small fraction of water-reactor fuel rods develop defects while in the reactor. Recent ranges for LWR fuel rod failure rates are 0.03%-0.04% (BWR) (74) and 0.01%-0.3% (PWR) (75). Historical aspects of reactor induced fuel failures have been reported and discussed in the context of spent-fuel storage (76).

The types of reactor-induced cladding defects include pinholes, cracks, and small holes, and a few cases where sections of cladding are missing.

Cladding defects which occur in-reactor permit the expulsion of fuel rod gases to the reactor coolant. The hot coolant will enter larger defects and dissolve soluble fission products. Fission products will adsorb on fuel assembly surfaces of intact and defective fuel.

During fuel discharge, fission products in the reactor coolant mix into the fuel pool water; they also desorb from fuel assembly surfaces during storage. Crud particles also are released from fuel assemblies during fuel handling. Crud spallation is enhanced when fuel is shipped in dry casks, which results in elevated fuel rod temperatures. Procedures have been used to control radioactivity at spent fuel pools. These procedures are summarized in IV.D.4.5.

The large majority of defective fuel assemblies stored in U.S. pools do not require canning. Several case histories (77) suggest that radiation releases from cladding defects are small during pool storage. The gaseous inventory was expelled in-reactor; further evolution at pool temperatures is negligible (57). Soluble species near large defects were dissolved by the reactor coolant. Leaching from the relatively few areas of exposed pellets appears to be low at pool temperatures. Studies are under way on defective fuel to further quantify radiation releases from defects under pool storage conditions (78).

Water may enter the fuel rod through cladding defects. Although this has not caused problems at pool temperatures, possible consequences if the fuel is to be removed to dry storage will be evaluated.

IV.D.4.2.3 Incidence of Fuel Damage in Storage and Handling

A survey of fuel handling mishaps in U.S. pools over the period 1974 to 1976 revealed nine cases (79) where an abnormal event occurred during fuel handling operations.* Radiation release was indicated in only one of the nine cases. In that case, a BWR fuel assembly was dropped ~30 ft (9 m) through water, while being removed from the reactor core, impacting on another fuel assembly. There was no detectable immediate radiation release to water or air. When the assembly was moved there was a momentary airborne release. The reactor room was evacuated but was soon returned to normal operation. In a separate incident, a Swedish BWR assembly was dropped approximately 8 m in pool water without causing a detectable radiation release or any visually detectable damage to the fuel rods or assembly hardware (80).

Water reactor fuel shipments in the U.S. have not resulted in obvious cladding degradation or serious radiation releases. One case of substantial radiation release inside a shipping cask was reported at a French reprocessing plant (70, 81). In that case, four stainless-clad fuel assemblies were shipped in a dry cask from the Chooz (Belgium) reactor (PWR) to the La Hague reprocessing plant in France. The fuel was stored at the reactor pool between 2 and 3 years. Fuel discharged at the same time had developed defects during the reactor exposure. However, defects were not obvious on the four assemblies; otherwise, they would have been canned for shipment. When the cask atmosphere was tested upon arrival at La Hague, it had a relatively high radiation level, indicating that a release had occurred from one or more rods during shipment. The fuel was discharged to the plant fuel pool, temporarily increasing the radiation level in the pool to $\sim 0.1\,\mu$Ci/ml. The pool cleanup system gradually reduced the pool radiation levels to normal values.

*Since the survey, three additional cases of dropped assemblies have been identified, none of which resulted in measurable radiation releases. One of these occurred in 1974 and was not reported in the original survey. The other two occurred in 1977.

Investigations of nuclear fuel integrity in several countries provide the following conclusions (15, 78, 80-88):

1. No observations or investigations have yet shown evidence of even minor corrosion-induced degradation on commercial water reactor fuel in pool storage, including visual inspection, pool radiation monitoring, non-destructive and metallurgical investigations on spent fuel after periods of water storage.

2. In contrast, deterioration was observable on pool-stored Magnox* fuel, stainless-clad gas reactor fuel (sensitized during reactor exposures), and defective Zircaloy-clad metallic uranium fuel. Thus, cladding degradation or fuel failures are often observable when they occur.

   The evidences have included gas releases, radiation releases to pool water, fuel assembly visual appearance, and metallographic evidence.

3. Assessments of known water reactor spent fuel assembly materials behavior have not indicated probable degradation mechanisms having serious consequences (15, 78, 80, 82-85).

4. Even in view of apparent and expected favorable storage characteristics of water reactor spent fuel, surveillance programs are in progress to identify any problems which might develop (15, 71, 78, 82-86).

IV.D.4.3 <u>Programs to Investigate and Evaluate Integrity of Spent Fuel During Storage</u>

Although prior experience has provided a sound basis for believing that water pool storage offers a safe and effective method of storing spent nuc'ear fuel for extended periods of time, both United States and foreign laboratories are continuing to investigate factors pertinent to the longterm in..egrity of the fuel assemblies in water pool environments. Dry storage, though at present backed by a somewhat lesser fund of information and operational data, is also being investigated on a broad basis to provide further support for the conclusions regarding its adequacy and comparative advantages. The following paragraphs discuss these continuing activities.

---

*Magnox refers to a magnesium-based alloy used as fuel cladding.

IV.D.4.3.1 Water Pool Storage Investigations

Investigations are being conducted in several countries to determine effects of water storage on spent fuel. Some of these programs will include periodic surveillance of the fuel for as long as required, utilizing non-destructive and metallurgical methods.

IV.D.4.3.1.1 U.S. Department of Energy Fuel and Fuel Pool Component Integrity Program

Pacific Northwest Laboratory (PNL, operated for the Department by Battelle Memorial Institute) is conducting a program to investigate spent fuel integrity, including metallurgical examinations of Zircaloy-clad Shippingport fuel (20-year storage), stainless-clad PWR fuel, typical PWR Zircaloy-clad fuel, fuel with reactor-induced defects, and eventually high burnup Zircaloy-clad fuel from ongoing demonstration programs (70).

The first phase of the United States program centers on examination of fuel assemblies selected for long-term surveillance. The selected assemblies have generally had prior examinations to define effects of the reactor exposure. They will now be examined to determine whether deterioration has occurred during water storage. The assemblies will then be subject to periodic long-term surveillance by visual, non-destructive and destructive methods for as long as required, to determine whether significant deterioration is occurring. Once the Department builds or accepts custody of one or more AFR's, this nucleus of surveillance assemblies will be stored there. Additional assemblies in the AFR inventory will be identified for inspection by random selection, to supplement the "nucleus" surveillance program. If evidence were to develop that fuel assembly deterioration is occurring, the number of supplemental examinations would be expanded and perhaps directed to specific types of fuel.

Investigations of some aspects of fuel assembly behavior, such as fission product and crud desorption and behavior in transients are under way as planned in foreign programs. The U.S. program intends to complement and also to interact with the foreign programs, particularly BEFAST, (IV.D.4.3.1.2) to provide a data base which addresses both immediate and long-term needs of water storage technology.

Another important and current aspect of the Department program is to identify and address problem areas in the operation of spent-fuel pool components, such as piping, racks, liners, and heat exchangers. Recent and current action includes assessments of corrosion experience of spent-fuel pool materials under conditions present in water pools, from the literature, technical meetings, and discussions with specialists. Further effort has included acquiring and inspecting sections from spent-fuel pool components after substantial pool residence. Some aspects of this activity are summarized in IV.D.4.4.

IV.D.4.3.1.2 <u>Behavior of Fuel Assemblies in Storage (BEFAST) Program</u>

BEFAST is an international program, operating under the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development. It was organized to provide international cooperation and information exchange in the area of spent-fuel storage. BEFAST, in cooperation with the International Atomic Energy Agency, is conducting a broadly based survey of spent fuel storage experience (70). It also provides a forum for rapid dissemination of spent-fuel storage information.

Experimental aspects of the BEFAST program have been proposed. Because of its recent incorporation under NEA (November 1979), the prospects for cooperative programs remain to be explored in detail.

The Pacific Northwest Laboratory represents the Department of Energy on the BEFAST committee.

IV.D.4.3.1.3 <u>Canadian (Atomic Energy of Canada, Ltd) Program</u>

Canadian spent-fuel surveillance includes a program at Chalk River Nuclear Laboratories to periodically examine 140 Zircaloy-clad nuclear fuel rods (82). The first examinations were conducted in 1978 on spent fuel from several Canadian reactors. Some of the fuel had been stored since 1962 and 1963. Based on the absence of detectable cladding deterioration, the investigators suggested that water storage of the fuel over a period of 50 years should present no problem. The Canadian program presently calls for further fuel examinations on a 5-year schedule through 1995.

Other aspects of the Canadian program include investigations of potential degradation mechanisms under fuel storage conditions. Of particular interest is a study of the likelihood that fission product iodine would cause Zircaloy cladding to degrade on the interior surface. To date their studies suggest that cladding failure from this source is unlikely.

IV.D.4.3.1.4 Federal Republic of Germany (FRG)--Kraftwerk Union Program and WAK Program

Kraftwerk Union is conducting a spent fuel surveillance program which includes the following elements (78, 84, 89).

1. Non-destructive examination of PWR Zircaloy-clad fuel with a burnup of 39,000 MWd/MTU and about 4 years in-pool, followed by future periodic examination. Future metallurgical examinations are possible if non-destructive testing suggests that deterioration is occurring.

2. Periodic surveillance on ten Zircaloy-clad rods with reactor-induced defects. After 3 years in storage no change in defect size has been detected.

3. Determination of radiolysis and fuel defect effects.

At the WAK reprocessing demonstration plant at Karlsruhe, a PWR fuel assembly (29,000 MWd/MTU) is annually removed to a hot cell and inspected (84, 89). After 6 years there is no visual evidence of cladding deterioration. There was no evidence of defect deterioration or UO_2 leaching on a BWR fuel assembly (19,800 MWd/MTU) with a defective rod, after 5 years of water storage.

IV.D.4.3.1.5 United Kingdom Program

As part of the data base developed for the Windscale (United Kingdom) reprocessing hearings, the following spent fuel was examined metallurgically (83, 86):

1. A Canadian Zircaloy-clad fuel bundle (6,500 MWd/-MTU) was examined after 11 years in pool storage. There was no evidence of pool-induced corrosion or other degradation.

2. Three BWR Zircaloy-clad rods (20,000 MWd/MTU, 6 years in-pool) and three PWR rods (33,000 MWd/MTU, 5 years in-pool) were examined. Again, there was no evidence of pool-induced degradation.

3. A Zircaloy-clad Steam Generating Heavy Water Reactor (SGHWR) fuel assembly containing two failed rods was placed in a closed can after a burnup of 1,900 MWd/MTU. After 9 years, the radioactive materials contained in the water inside the can had risen to 1mCi (5 ppm of ^{137}Cs). A detailed hot-cell examination of the defective fuel rods indicated only small increases in fuel rod diameter at the defects, with no evidence that a UO_2 - U_3O_8 conversion was occurring. There was no evidence of poolinduced degradation on the Zircaloy cladding or on the stain-less-steel spacer. Some mild corrosion of ferritic steel mandrels had occurred. Non-destructive examinations also were performed on fuel rods from the assemblies described above.

IV.D.4.3.1.6 Summary of Results of Water Pool Storage Investigations

In summary, there is no evidence, either by visual observations, by radiation monitoring of pool air and water, or by metallurgical or non-destructive examinations cited above, that Zircaloy-clad water reactor fuel is degrading in pool storage, including fuel with up to 20 years of pool residence.

Further evidence of excellent aqueous corrosion resistance of irradiated Zircaloy-clad fuel comes from the following cases:

1. Canadian Zircaloy-clad assemblies were returned to a reactor after 10, 9, and 5 years, respectively, in pool storage (15). The fuel performed satisfactorily at power ratings near those in the Pickering reactor (4.2 kW/m).

2. Eight Zircaloy-clad fuel assemblies loaded into the NPD reactor in 1963 are still operating satisfactorily at the reactor primary conditions (90). Two assemblies examined after 3,000 hours had developed only 3 to 5 μm of oxide.

3. Another Zircaloy-clad fuel assembly examined after 17 years in the Shippingport reactor (burnup of 40,000 MWd/MTU) had developed ~20 μm (maximum) of oxide and no other evidence of degradation (87).

One stainless-clad PWR rod was examined for the Windscale hearings without seeing evidence of degradation after 3 years in-reactor and 5 years in-pool. Only two United States PWR's and one small BWR use stainless-clad fuel. While the fractional inventory of stored stainless-clad PWR fuel is relatively small (less than 10% of the U.S. spent-fuel inventory), the number of U.S. assemblies was 1,475 as of mid-1979, appearing to justify some additional surveillance under the United States program. Approximately 850 stainless-clad LWR assemblies were stored in foreign spent fuel pools as of mid-1979.

Under the Department of Energy Fuel Integrity Program at PNL, negotiations are in an advanced stage to acquire a stainless-clad PWR fuel assembly for near-term and long-term surveillance. The negotiations also include access to information from two other stainless-clad fuel assemblies which are scheduled for examination in the next few months.

The above evidence is positive, but preliminary. It will be augmented by spent-fuel surveillance activities which are under way in several countries as outlined above.

IV.D.4.3.2 Dry Storage Investigations

The Canadian spent fuel program has included a dry storage demonstration at the Atomic Energy of Canada Whiteshell Nuclear Laboratories. One hundred thirty-eight irradiated Zircaloy-clad fuel assemblies from the WR-1 and Douglas Point reactors were encapsulated in steel canisters and placed in an air storage demonstration test in late 1975. Another 360 spent fuel assemblies from the Douglas Point reactor were placed in air storage beginning in mid-1976 (85). The encapsulated assemblies are operating successfully.

In connection with the Karnbranslesakerhet (KBS) study of the disposal of spent fuel, Swedish investigators have developed both theoretical and experimental concepts for encapsulating spent fuel for final disposal under dry conditions (88).

U.S. programs have included evaluation of dry storage concepts using electrically heated fuel assembly simulators (31). More recently, spent-fuel assemblies from the Turkey Point reactor (PWR) have been placed in dry storage demonstration modules (91). Current assessments suggest that dry storage of spent fuel can be accomplished successfully.

In summary, there is existing technology to encapsulate defective spent fuel for storage in spent-fuel pools if required. There also is a preliminary, but substantial data base emerging for fuel encapsulation for dry storage. Either concept provides a backup position if unexpected deterioration were to develop on pool-stored fuel.

IV.D.4.4 Possible Impacts of Other Research and Development

Behavior of spent-fuel components is a secondary, though important consideration in spent-fuel management. In 35 years of experience, storage racks, pool liners, grapples, etc., have presented minimal operational problems.

Another consideration relevant to spent fuel storage is the behavior of fuel during reactor residence. Extended burnup is being investigated to improve uranium utilization. Also, programs are in progress to reduce the incidence of reactor-induced fuel failures. Programs relating to spent fuel component behavior and in-reactor behavior of nuclear fuel are indicated below.

IV.D.4.4.1 Programs Relevant to Spent-Fuel Pool Component Integrity (Racks, Liners, and Piping)

The Department's program includes investigation of spent fuel pool component behavior (70). Under current investigation is an intergranular cracking phenomenon which developed at field welds in Type-304 stainless steel piping in PWR spent-fuel pools. Investigations of the cracking phenomenon also are being conducted under programs sponsored by the Electric Power Research Institute.

Current definition of the cracking phenomenon is summarized below:

1. The cracks develop in weld-heat affected zones of 304 stainless steel pipes in spent fuel pools.

2. The cracks propagate from the inside (coolant) surface by an intergranular mechanism.

3. The cracking appears to be confined to pipes where the boric acid solution was stagnant over extended periods.

4. The cracking appears to occur in pipes with relatively high carbon levels (≥0.065 wt. % C), where sensitization during weld heat cycles is most likely to occur.

The question of whether similar cracking might occur on welded 304 stainless steel fuel storage racks was explored at the Zion Reactor rerack hearing. An electrochemical inspection of welds on the Zion racks suggested that weld sensitization was below levels which have been found to promote intergranular stress corrosion cracking.

Two Type-304 stainless steel rack sections, exposed to boric acid chemistries in PWR pools, were examined metallographically at PNL (70).

A section from a standard shop-welded rack had no evidence of corrosion attack after 6-2/3 yr in a boric acid pool. A poorly made weld on a rack stand had a shallow (0.001 to 0.003 in. deep) intergranular attack after 1-1/2 yr in a PWR pool. However, there was no evidence of stress corrosion cracking.

The PWR pipe crack problem appears to have marked similarities to cracking which has developed in Type-304 stainless steel pipes in BWR's at higher temperatures. The parallels between the two cases are being explored. Much information regarding the cracking phenomenon and how to avoid it may already be available from BWR studies.

To this point, the consequence of the PWR spent-fuel pool pipe cracking is that some pipe sections have been replaced (with less-sensitization-prone Type-304L stainless steel).

Studies sponsored by the Department program and by the Electric Power Research Institute are under way to better define the cracking mechanism, including the important factors. The studies include Constant Extension Rate Testing (CERT) to investigate combined effects of stress and environment under controlled conditions. Also, probes are being installed in spent-fuel pool piping at a PWR to monitor possible corrosion-inducing conditions in stagnant and flowing coolant. The programs planned and in progress are expected to provide a basis for recommendations to fuel pool designers and operators which will allow selection of materials and operating regimes to avoid the cracking problem.

Fuel component investigations also are being conducted under foreign programs. The KWU program has included examination of a fuel handling machine exposed to boric acid pool chemistry for 12 years. Complete disassembly indicated that the stainless steel components were still in excellent condition and they were reused. Investigation of fuel rack material corrosion at the WAK pool (89), Karlsruhe, Germany, indicated satisfactory behavior.

IV.D.4.4.2 Programs Relevant to Definition of In-Reactor Spent-Fuel Behavior

Several studies are under way in the United States and elsewhere to investigate extending burnups on LWR fuel into the range of 40,000 to 60,000 MWd/MTU. Mentioned earlier was a joint Spanish-U.S. program which successfully irradiated Zircaloy-clad fuel rods to 62,000 MWd/MTU (71).

Other high-burnup and fuel improvement programs are being sponsored in the United States by the Department (67) and the Electric Power Research Institute (69) in cooperation with fuel vendors, utilities, and research laboratories. Maximum burnups now projected for near-term demonstration fuel are 55,000 MWd/MTU. The Department's Spent-Fuel and Fuel Component Integrity Program will closely follow results of high-burnup fuel examinations for developments relevant to spent-fuel storage. Plans are under way to eventually accept custody of one or more high-burnup fuel assemblies for long-term surveillance which would define storage characteristics of highburnup fuel. Unusual storage behavior of high-burnup Zircaloy-clad fuel is not anticipated. However, the surveillance activities will provide a basis to assess highburnup fuel storage behavior before significant inventory of highburnup fuel develops.

Several programs are in progress in this country and elsewhere to improve performance of nuclear fuel, including suppression of fuel failure mechanisms (92). Failed fuel contributes to fission product inventories in spent-fuel pools, so minimizing in-reactor fuel failures has benefits for spent-fuel pool operations.

IV.D.4.4.3 Materials Studies Relevant to Long-Term Integrity of Spent-Fuel Cladding and Other Spent-Fuel Pool Materials

Many organizations such as universities and research laboratories conduct basic corrosion studies. Numerous studies are relevant to spent-fuel pool regimes and materials, but few studies have been intentionally directed to spent fuel pool technology because the water storage environments have caused very few significant corrosion problems. Recently some university and research laboratory support has developed to gain further understanding of stainless steel pipe cracking in PWR spent-fuel pools.

Corrosion has not been a substantial source of problems over the 35 years of pool storage experience. However, it now seems appropriate to direct some research capability to investigate and possibly anticipate corrosion problems in this important and rapidly expanding area of nuclear technology.

IV.D.4.5 Surveillance of Fuel in Storage and Corrective Action

Monitoring of spent fuel in storage is a significant aspect of developing continuing assurance of safety of extended (40-50 years) water storage. Monitoring includes visual inspections, radioactivity monitoring, pool chemistry, and non-destructive and destructive examinations of fuel and fuel-pool components.

There are existing procedures to deal with routine and nonroutine releases of radioactive species in spent-fuel pools.

IV.D.4.5.1 Methods to Monitor Spent-Fuel Cladding and Other Pool Materials

Spent fuel integrity is monitored by non-destructive (93) and destructive examinations (see IV.D.4.3) to assess physical, chemical, and mechanical characteristics. Specific problems that have arisen with fuel in operating light water reactors (e.g., hydriding of Zircaloy cladding, fuel densification, oxide formation, fuel pellet-cladding interaction) were identified by non-destructive examinations at spent-fuel pools. Areas of interest during non-destructive examinations include (i) indications of possible areas of cladding degradation and (ii) the behavior of defective fuel. Destructive examinations are performed to thoroughly characterize the condition of spent fuel and to supplement and/or verify the results from the non-destructive examinations. Typical non-destructive inspection techniques are listed in Table IV-14.

Table IV-14. Inspection Techniques Involved in Non-Destructive Examination of Spent Fuel

Inspection Technique	Inspection Techniques Used On: Fuel Assembly	Fuel Rod
Visual:		
Optical	x	x
Photography	x	
Television	x	x
Dimensional:		
Profile		x
Gaps	x	x
Lengths/widths	x	x
Weighing	x	x
Leak testing	x[a]	x[b]
Gamma scan	x[c]	x
Eddy current		x
Ultrasonic		x
Gamma flux	x	
Neutron flux	x	

[a]By sipping methods (wet, dry, hybrid, vacuum).

[b]By sipping or other method (e.g., helium pressurization).

[c]Sometimes corner rods are gamma scanned while the rods are still in the fuel assembly.

Destructive examinations are generally performed on fuel rods in hot cell facilities. Typical inspection techniques involved in destructive examinations of irradiated fuel are listed in Table IV-15.

Non-destructive and destructive methods also are applied to monitor the condition of spent-fuel pool components. Ultrasonic non-destructive testing has been applied to identify actual and incipient cracks in spent-fuel pool piping. Sections from spent-fuel pool components have been investigated by metallography for evidence of corrosion (70). Both nondestructive and destructive (metallurgical) monitoring procedures will continue to be applied to spent-fuel and pool components.

Table IV-15. Inspection Techniques Involved in Destructive Examination of Spent Fuel

Visual (optical/photography)
Dimensional:

- Lengths/widths
- Profilometry

Fission gas analysis:

- Composition
- Percent release
- Fuel rod internal pressure

Metallography of cladding

Ceramography of fuel

Autoradiography (alpha and beta-gamma)

Cladding hydrogen analysis

Electron microscopy

Actinide assay

Burnup analysis

Cladding mechanical property testing

Leaching rates of fuel and fission products from exposed fuel

IV.D.4.5.2 Procedures to Handle Radiation Releases in Spent-Fuel Pools

Both dissolved and particulate radioactive species are released to spent fuel pool water, either from the fuel surfaces or from mixing of reactor pool and fuel pool waters. Methods are available to minimize the consequences. Small amounts of airborne activity are released from the pool water surface, particularly during refueling at reactor pools and when pool temperatures rise substantially. Dissolved and particulate species are released from spent fuel during handling and storage.

Routine procedures are applied to minimize both dissolved and particulate species in the pool water:

1. Ion exchange to remove ionic species.
2. Filters to remove particulate species.
3. Skimmers to remove material from the pool surface.
4. Vacuum cleaners to remove particles from the pool floor.

These procedures result in routine radioactivity concentrations of 10^{-3} μCi/ml at reactor pools except during fuel discharges and 10^{-4} to 10^{-5} μCi/ml at AFR's.

Gas hoods have been devised to channel gases away from the pool staff if a cladding failure were to develop in the pool. To date they have not been needed for water reactor fuel, because in-pool failures have not materialized.

IV.D.4.5.3 Alternative Procedures if Problems Eventually Were to Develop With Fuel in Water Storage

Although there is no evidence available so far to indicate that backup provisions will be required for water pool storage, such backup is available if necessary. Encapsulation of spent fuel prior to introduction to storage has been practiced and information is available concerning the behavior of the encapsulating material in the water pool environment.

In the case of in-reactor defects, operators have found that the fuel stores satisfactory without encapsulation (77). This is due to the fact that the gaseous radioactive inventory has discharged in-reactor. Pinhole defects tend to shrink or close on cooling. Leach rates are relatively low at the small number of defects where pellets are exposed.

If cladding failures were to occur on pool-stored fuel, need for encapsulation would be a major consideration, though not a foregone conclusion. The need would depend on the failure types, the species, and the amounts of isotopes released, and the number of failures which were anticipated. In-pool failures have occurred on Magnox fuel, stainless-clad gas reactor fuel (86), and on defective Zircaloy-clad metallic uranium fuel (15), without preempting pool operation. Swedish investigators intentionally drilled into a fuel rod to determine the amount of radioactive gas which would be released if a fuel rod failed during underwater handling operations. The actual gas release was extremely small (80).

Encapsulation isolates the fuel from the pool water. The encapsulation can either be dry-in-wet (fuel dry inside a pool-stored can) or wet-in-wet (fuel wet inside pool-stored can). Basic studies to confirm long term adequacy of the procedure, such as radiolytic effects inside the wet-in-wet cans, are in progress (78). Defective Zircaloy-clad fuel showed no unusual behavior after 9 years inside a water-filled can (83).

Materials studies also are being directed to final disposal of encapsulated spent fuel. Some individual spent fuel rods have been placed in air storage, including relatively fresh rods with burnups up to 62,000 MWd/MTU (Zorita fuel stored in Spain). Entire bundles have been stored dry for time spans of months or years in the United States and Canada without problems.

The need for a large-scale emergency encapsulation program for dry storage of spent fuel seems remote. If degradation were to develop it seems likely to do so slowly under the low temperatures characteristics of fuel pools, providing time to identify and implement a rational response.

IV.E SUMMARY OF TECHNICAL BASIS FOR CONFIDENCE THAT SPENT FUEL CAN BE STORED IN A SAFE AND ENVIRONMENTALLY ACCEPTABLE MANNER

The preceding discussion in Part IV has described the technology of water pool storage of spent fuel, has reviewed the performance requirements of water pool facilities and how these requirements are met, and has discussed the background of information on operational experience with water pool storage. These considerations are basic to a principal issue in this proceeding: Can spent fuel be stored safely and in an environmentally acceptable manner, and if so, can such storage be safely extended until a respository is available? The abundant evidence cited in IV.B, IV.C, and IV.D confirms the adequacy and safety of extended storage of spent fuel in a water pool environment. The following points summarize the basis of this conclusion:

1. The technology of water pool storage of spent fuel is not only available but is well established through more than 30 years of work at government and industrial facilities. Dry storage of spent fuel by several different techniques has been the subject of a significant level of research, development, and demonstration, and promises to be a technically viable alternative to water pool storage. Thus, there are a number of technically suitable alternative methods of spent-fuel storage in existence at the present time.

2. The regulatory framework, industry standards, and design requirements for the water pool storage of spent fuel currently exist.

3. The licensing of water pool storage of spent fuel has been practiced routinely by the Commission and its predecessor agency for nearly 20 years and is being practiced at the present time.

4. Zircaloy-clad spent fuel has been stored under water for periods up to 20 years, and stainless steel-clad fuel has been so stored for periods up to 12 years, with no evidence of degradation as a result of such storage. Studies of the corrosion aspects of water pool storage indicate that there are no obvious degradation mechanisms which operate on the cladding at rates which would be expected to cause failure in the time frame of 50 years or longer. Moreover, in the unlikely event that severe deterioration of the cladding were to develop, the spent fuel could be encapsulated to provide the necessary integrity for indefinite storage.

IV.E.1 Storage in At-Reactor Facilities

Because much of the experience in handling and storage of spent fuel has been gained at reactor sites and much of the technical data discussed in this part was acquired in studies at reactor storage pools, there is no reason to doubt the technical adequacy of existing and planned reactor storage pools. For this reason, continued storage of spent fuel at reactor sites would be acceptable, even if such storage should be required for a period beyond the time of expiration of the reactor operating license. This conclusion is based on the following considerations:

1. Water basin storage at reactor sites involves the same kinds of considerations as those set forth above with respect to

    a. Performance of fuel under extended storage conditions.

    b. The benign character of the storage activity relative to radioactivity releases, radiation exposures to plant workers and the public, accident evaluations, and other safety aspects.

    c. Waste generation during the course of storage.

    d. Facility requirements.

2. Facilities in existence and those under construction at reactor sites are designed and constructed to more rigorous standards than would be required under proposed 10 CFR 72.

3. The environmental impact of continued storage of spent fuel at reactor sites has been evaluated by the Nuclear Regulatory Commission (1) and found to be acceptable. No time-dependent factors adversely affecting long-term safety of such storage have been identified.

4. Although there is some interdependence between the spent fuel-pool and reactor operation at reactor sites, this interdependence is limited to the supply of utilities and waste management services. The fuel storage operation depends on the reactor plant utility system for steam, cooling water, deionized water, and handling of low-level wastes, as well as for certain personnel services such as health physics, safety, and personnel. All of these could be continued relatively easily following shutdown of the reactor.

5. There is no technical reason why spent fuel cannot be stored at the reactor site after reactor operation ceases.

6. Such continued storage would remain under NRC licensing authority.

IV.E.2 Storage in Away-From-Reactor Facilities

All points cited above regarding the confidence in water pool storage generally are applicable to AFR storage systems. Therefore, there is sufficient technical information available that AFR spent-fuel storage facilities can be built and operated in a manner which is both safe and environmentally acceptable, and which otherwise meet the goals set forth in IV.A. The information presented in IV.B, IV.C, and IV.D and summarized above demonstrates that large AFR spent-fuel storage facilities can be constructed and operated to meet necessary safety requirements with a minimum impact on the environment, and in compliance with applicable regulations.

References for Part IV

(1) U.S. Nuclear Regulatory Commission, Generic Environmental Impact Statement - Handling and Storage of Spent LWR Fuel, NUREG-0575, August 1979

(2) 10 CFR, Chapter I, Part 20, "Standards for Protection Against Radiation"

(3) 10 CFR, Chapter I, Part 100, "Reactor Site Criteria"

(4) 10 CFR, Chapter I, Part 72 (Proposed), "Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," 43 Fed. Reg. 46309, (October 1978)

(5) Allied General Nuclear Services, Barnwell Nuclear Fuel Plant-Safety Analysis Report, Submitted to U.S. Atomic Energy Commission, NRC Docket 50-332, 1969, as amended

(6) General Electric Company, Design and Analysis Report - Midwest Fuel Reprocessing Plant, NRC Docket 50-268, 1966, as amended

(7) Nuclear Fuel Services Inc., Safety Analysis Report - NFS Reprocessing Plant, West Valley, New York, NRC Docket 50-201, 1963, as amended

(8) E.I. du Pont de Nemours and Company, Spent Fuel Handling and Storage Facility for the International Spent Fuel Storage Program, DPSTD-ISFS-78-7, Savannah River Laboratory, Aiken, SC, April 1978

(9) E.I. du Pont de Nemours and Company, Conceptual Design Report, The Away From Reactor Spent Fuel Storage Facility, DPE-3547, Savannah River Laboratory, Aiken, SC, December 1978

(10) Ibid. Section VII, Sheet 1.

(11) American Nuclear Society, "World List of Nuclear Power Plants," Nuclear News 23: No. 2, pp. 67ff, February 1980

(12) U.S. Nuclear Regulatory Commission, Draft Generic Environmental Impact Statement, Handling and Storage of Spent LWR Fuel, NUREG-0404, Vol. I, p. ES-5, March 1978

(13) L.V. Gossick, Testimony before the House Subcommittee on Oversight and Investigation, pp. 6-8, Committee on Interstate and Foreign Commerce, United States Congress, August 1977

(14) U.S. Energy Research and Development Administration, LWR Spent Fuel Disposition Capabilities, 1977-1986, p. 4, ERDA 77-25, May 1977

(15) A.B. Johnson Jr., Behavior of Spent Nuclear Fuel in Water Pool Storage, BNWL-2256, Battelle Northwest Laboratories, Richland, WA, 1977

(16) Nuclear Engineering International, "The World's Reactors, No 64, Fulton Generating Station, HTGR" 19: 635, p. 643, (Application for Construction Permit docketed November 1973), August 1974

(17) Delmarva Power and Light Company Preliminary Safety Analysis Report - Summit Nuclear Power Station, p. 1.2-1.12, Submitted to U.S. Atomic Energy Commission, NRC Docket 50-450, April 1973

(18) U.S. Nuclear Regulatory Commission, Facilities License Application Record, License Docket 50-171

(19) U.S. Energy Research and Development Administration, Waste Management Operations, Idaho National Engineering Laboratory, ERDA-1536, September 1977

(20) J.D. Hammond, R.S. P'Pool, and R. D. Modrow, Safety Analysis, Peach Bottom Spent Fuel Storage, IN-1465, Idaho National Engineering Laboratory, Idaho Falls, ID

(21) U.S. Energy Research and Development Administration, Environmental Impact Statement-Receipt, Storage, and Processing of Rover Fuel, WASH 1512, April 1972

(22) U.S. Nuclear Regulatory Commission, Facilities License Application Record, License Docket 50-267

(23) U.S. Energy Research and Development Administration, Draft Environmental Statement-HTGR Fuels Reprocessing Facilities, National Reactor Testing Station WASH-1534 January 1974

(24) R.J. Steffen, D. Ourrill, and J.B. Wright, "EMAD Support of NWTS Experiments and Demonstrations," Proceedings of the National Waste Terminal Storage Program Information Meeting, p. 154, ONWI-62, Battelle Memorial Institute, Columbus, OH, October 1979

(25) L.F. Richards and M. Szulinski, Subsurface Storage of Commercial Spent Nuclear Fuel, p. VI-20, CONF-780304, ANS Conference, Savannah, GA, March 1978

(26) J.A. Morrison, AECL Experience in Managing Radioactive Wastes from Canadian Nuclear Reactors AECL-4707, Atomic Energy of Canada, Limited, Chalk River, Ontario (March 1974); Quoted in ERDA-76-43, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the Nuclear Fuel Cycle, Vol. 3, May 1976

(27) K.J. Truss, Concrete Canister Demonstration Project, Preliminary Safety Analysis Report, Whiteshell Nuclear Research Establishment, Pinaua, Manitoba, 1975; quoted in ERDA-76-43, Alternatives for Managing Wastes from Reactors and Post Fission Operations in the Nuclear Fuel Cycle Vol. 3, 1976

(28) P.J. Dyne, Managing Nuclear Wastes AECL-5136, Atomic Energy of Canada, Limited, Chalk River, Ontario, May 1975. Quoted in ERDA-76-43, Alternatives for Managing Wastes from Reactors and Post Fission Operations in the Nuclear Fuel Cycle, Vol. 3, 1976

(29) Atlantic Richfield Hanford Co. and Kaiser Engineers, Retrievable Surface Storage Facility Alternative Concepts Engineering Studies, ARH-3100, Atlantic Richfield Hanford Company, Richland, WA, July 1974

(30) J.R. La Riviere, and D.C. Nelson, "Retrievable Surface Storage: Interim Storage of Solidified High Level Waste," CONF-760701, Proceedings of the International Symposium on the Management of Wastes From the LWR Fuel Cycle, Denver, CO, July 1976

(31) J.M. Davis, Demonstration of a Surface Storage System for Spent Fuel or Waste, CONF-771102-18, Atlantic Richfield Hanford Co., Presented at 20th Annual Meeting, American Institute of Chemical Engineers, New York City, NY, November 1977

(32) K.H. Henry and D.A. Turner, Storage of Spent Unreprocessed Fuel (SURF), CONF-780316-5, Rockwell Hanford Operations, Richland, WA, March 1978

(33) Nucleonics Week, Special Supplement, International Nuclear Fuel Cycle Evaluation, p. S-6, 21 February 1980

(34) International Atomic Energy Agency, Storage, Handling, and Movement of Fuel and Related Components at Nuclear Power Plants, Technical Report Series No. 189, (1979) p. 42.

(35) 10 CFR, Chapter I, Part 30, "Rules of General Applicability to Licensing of By-Product Material"

(36) 10 CFR, Chapter I, Part 40, "Licensing of Source Material"

(37) 10 CFR, Chapter I, Part 70, "Special Nuclear Material"

(38) 10 CFR, Chapter I, Part 50, "Licensing of Production of Utilization Facilities"

(39) U.S. Nuclear Regulatory Commission, Regulatory Guide 3.44, "Standard Format and Content for the Safety Analysis Report To Be Included in a License Application for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (Water-Basin Type), December 1978

(40) U.S. Nuclear Regulatory Commission, Draft Guide 3.XX, "Guidance on the License Application, Siting, Design, and Plant Protection for an Independent Spent Fuel Storage Installation," November 1974

(41) U.S. Nuclear Regulatory Commission, Regulatory Guide 3.24, "Guidance on the License Application, Siting, Design, and Plant Protection for an Independent Spent Fuel Storage Installation," December 1974

(42) U.S. Nuclear Regulatory Commission, Regulatory Guide 1.13, "Spent Fuel Storage Design Basis," Revision 1, 1975

(43) 40 CFR, Chapter 1, Subchapter F, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," 42 Fed. Reg. 2858 (January 1977)

(44) American Nuclear Society/American National Standards Institute, Draft Standard N57.7, Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type), 1978

(45) American National Standards Institute, Draft Standard N-XXX, Guidelines for Evaluating Site-Related Parameters for an Independent Spent Fuel Storage Installation (ANS-2.19), September 1978

(46) U.S. Nuclear Regulatory Commission, Generic Environmental Impact Statement - Handling and Storage of Spent LWR Fuel, NUREG-0575, Vol. 1, p. ES-10, August 1979

(47) U.S. Department of Energy, Draft Environmental Impact Statement - Storage of U.S. Spent Power Reactor Fuel, DOE/EIS-0015D, August 1978

(48) U.S. Nuclear Regulatory Commission, Facilities License Application Record, License Docket 50-201, License CSF-1

(49) U.S. Energy Research and Development Administration, LWR Spent Fuel Disposition Capabilities, 1975-1984, Table II-3, Savannah River Operations Office, March 1975

(50) U.S. Nuclear Regulatory Commission, Facilities License Application Record, License Docket 50-268, and 10 CFR 70, License No. SNM-1265

(51) U.S. Nuclear Regulatory Commission, Mixed Oxide Fuel - Order, December 23, 1977, Docketed December 27, 1977

(52) U.S. Nuclear Regulatory Commission, Facilities License Application Record, NRC Docket 50-10, License No. DPR-2 (Dresden Nuclear Power Station, Unit 1)

(53) U.S. Nuclear Regulatory Commission, Facilities License Application Record, December 1979

(54) U.S. Department of Energy, Draft Environmental Impact Statement - Storage of U.S. Spent Power Reactor Fuel, Supplement, p. S-I.4, DOE/EIS-0015DS, December 1978

(55) L.R. Haywood, J.A.L. Robertson, J. Pawliw, J. Howieson, and L.L. Bodie, Fuel for Canadian Power Reactors, p. 187, CONF 49/A/156, September 1971

(56) R.D. Page, "Fuel for CANDU Pressurized Heavy Water Reactors," Nuclear Engineering International, Vol. 19, No. 217, p. 496, June 1974

(57) A.B. Johnson, Jr., "Spent Fuel Storage Experience," Nucl. Tech., Vol. 43, pp. 165-173, 1979

(58) W.H. Baker and F.D. King, Technical Data Summary, Spent Fuel Handling and Storage Facility for LWR Fuel Reprocessing Plant DPSTD-AFCT-77-7, Savannah River Laboratory, Aiken, SC, August 1977

(59) U.S. Department of Energy, Analytical Methodology and Facility Description - Spent Fuel Policy, DOE/ET-0054, August 1978

(60) Ibid., p. I-8

(61) Ibid., p. I-10

(62) Ibid., P. I-11

(63) M.J. Bell, ORIGEN - The ORNL Isotope Generation and Depletion Code, USAEC Report ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, TN, 1973

(64) S.E. Turner et al., Historical Survey of Nuclear Fuel Utilization in U.S. LWR Power Plant, pp. 2 & 18, SSA-122, Southern Science Applications, Inc., 1979

(65) P. A. Aucoin (NAC), Memorandum dated 20 February 1980 to W. Bailey (PNL), "Data Source: Fuel Trac, Nuclear Assurance Coporation--correction to NAC-23, p. 29, January 1973,".

(66) P. A. Aucoin (NAC), Memorandum dated 20 February 1980 to W. Bailey (PNL), "Data Source: Fuel Trac, Nuclear Assurance Corporation--correction to NAC-25, pp. 24-25, January 1973

(67) P.M. Lang, "Future Trends in LWR Fuel to Improve Uranium Utilization," Trans. Am. Nuclear Soc., Vol. 30, pp. 274-275, November 1978

(68) M. Levenson, "Incentives for Extending Fuel Burnup," EPRI Journal, pp. 47-49, December 1978

(69) J.T.A. Roberts et al., LWR Fuel Performance Program: Progress in 1978, EPRI NP-1024SR, Section 3, p. 30, February 1979

(70) A.B. Johnson, Jr., et al., Annual Report - FY 1979, Spent Fuel and Fuel Pool Component Integrity, PNL-3171, Pacific Northwest Laboratory, Richland, WA, 1979 in press

(71) E. Roberts et al., "Fuel Modeling and Performance of High Burnup Fuel Rods" ANS Topical Meeting on Water Reactor Fuel Performance, pp. 133-135, St. Charles, IL, May 1977

(72) F. Garzarolli et al., "The Main Causes of Fuel Element Failure in Water Cooled Power Reactors," Atomic Energy Review, Vol. 17(1), pp. 31-128, 1979.

(73) Handling and Final Storage of Unreprocessed Spent Nuclear Fuel, Vol 1., pp. 8, 9, 26, Karnbransksakerhet (KBS), Stockholm, Sweden, 1978

(74) W.E. Baily et al., Proc. ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, p. 8, April 29 to May 3, 1979

(75) M.G. Andrews et al., Proc. ANS Topical Meeting on Light Water Reactor Fuel Performance, p. 17, Portland, Oregon, April 29 to May 3, 1979

(76) C.W. Funk and L.D. Jacobson, Inventory and Characteristics of Spent Light Water Reactor Fuel, HEDL-TME-77-82, April 1979

(77) A.B. Johnson, Jr., "Impacts of Reactor-Induced Defects on Spent Fuel Storage," Proc. NEA Sem. Storage of Spent Fuel Elements, Madrid, Spain, June 20-23, 1978

(78) M. Peehs et al., "Behavior of Spent LWR Fuel Assemblies," Proc. NEA Sem. Storage of Spent Fuel Elements, p. 231, Madrid, Spain, June 20-23, 1978

(79) A.B. Johnson Jr., "Behavior of Spent Nuclear Fuel in Water Pool Storage," p. 25-26, BNWL-2256, Battelle Northwest Laboratories, Richland, WA, 1977

(80) G. Vesterlund and T. Olsson, Degraderingmeckanismer Vid Bassanglagering och Hantering av Utbrant Kraftreaktorbransle, RU 78-29, AESA-ATOM, Vasteras, Sweden. English Translation, BNWL-TR-320, p. 40, Battelle Northwest Laboratories, May 1978

(81) G. LeFort, and J.C. Puit, "La Surete des Installations Francaises de Stockage des Elements Combustibles Irradies de la Filliere de Eau Figere," Proc NEA Sem. Storage of Spent Fuel Elements, p. 106, Madrid, Spain, June 20-23, 1978, and discussion section. Conf. Proc.

(82) C.E.L. Hunt et al., Long-Term Storage of Fuel in Water, AECL-6577, Atomic Energy of Canada, Limited, Chalk River, Ontario, 1979

(83) R.H. Flowers, Results of an Examination of Irradiated Oxide Fuel Following Storage in Water, Testimony Presented at Windscale Inquiry on Spent Fuel Reprocessing, Windscale, U.K., October 1977

(84) K.L. Huppert, "Spent Fuel Storage - Philosophies and Experience," Proc. NEA Sem. Storage of Spent Fuel Elements, Madrid, Spain, June 20-23, 1978

(85) S.A. Mayman, "Canadian Experience with Wet and Dry Storage Concepts," ANS Executive Conference Spent Fuel Policy and Its Implications, Buford, GA, April 1978

(86) Honorable Mr. Justice Parker, The Windscale Inquiry, Vol. 1, p. 6., Report and Annexes 3-5, Her Majesty's Stationery Office, London, England, Jan. 26, 1978

(87) E. Hillner, Corrosion and Hydriding Performance of Zircaloy Tubing after Extended Exposure in the Shippingport Pressurized Water Reactor, ASTM STP-551, pp. 449-462, 1974, see also WAPD-TM-1412

(88) Handling and Final Storage of Unreprocessed Spent Nuclear Fuel, Vol. 2, Karnbranslesakerhet (KBS), Stockholm, Sweden, 1978

(89) K.L. Huppert and G. Zimmerman, Experience with Regard to Long-Term Storage of LWR Fuel Elements, pp. 447-450 (in German), Reactor Congress, Mannheim, AED-CONF-77-013- 112, March 29-April 1, 1977

(90) A.B. Johnson Jr., Behavior of Spent Nuclear Fuel in Water Pool Storage, p. 15, BNWL-2256, Battelle Northwest Laboratories. Richland, WA, 1977

(91) L.D. Ramspott et al., Technical Concept for a Test of Geologic Storage of Spent Reactor Fuel in the Climax Granite, Nevada Test Site, pp. 18-19, UCRL-52796, Lawrence Livermore National Laboratory, Livermore, CA, June 15, 1979

(92) Proceedings of ANS Topical Meeting on Light Water Reactor Fuel Performance, pp. 59-67 & 74-112, Portland, OR, April 29 to May 3, 1979

(93) S.Y. Ogawa et al., "Techniques and Equipment for Fuel Examinations at Power Reactor Sites," Proceedings of 21st Conference on Remote Systems Technology, pp. 104-108, 1973

V PROGRAM FOR PROVIDING AWAY-FROM-REACTOR STORAGE FACILITIES

This part describes the program which has been developed and is being implemented by the Department of Energy in order to be able to receive and store on a timely basis spent fuel from commercial reactors in government-owned or controlled away-from-reactor (AFR) storage facilities. This management plan includes consideration of:

1. AFR storage requirements.
2. Institutional issues.
3. Method of implementation of government AFR projects and the schedule therefor.
4. Costs associated with the acquisition and construction of AFR storage facilities.

In 1977, the Department initiated a program to determine the amount of storage capacity that might be required and to determine the best method for meeting such requirements (1). The President's statement of 12 February 1980, establishing a comprehensive radioactive waste management program (2), emphasized that the storage of commercial spent fuel is primarily the responsibility of the electric utilities, but that a limited amount of government storage capacity would provide an alternative to those utilities which are unable to expand their storage capabilities.

In December 1977, the Department requested from industry expressions of interest in providing spent-fuel storage services to the Department. After reviewing these responses, it was concluded that commercial storage services were not reasonably available without guarantees and commitments equivalent to those provided by a government-owned facility. The Department thus decided that it would either acquire or construct its own storage facilities. The Department has conducted a preliminary review (3) of 19 government and 3 private storage facilities. A conceptual design and site survey of 50 government sites as possible locations for a new AFR storage facility also were completed. After comparing storage requirements with facility development schedules, however, it was clear that only existing facilities could meet near-term requirements, because a new storage facility could not be designed, licensed, constructed, and made available until approximately 1989. Based on these studies, the following three locations were identified as options for potential near-term use:

1. General Electric Midwest Fuel Recovery Plant, Morris, Illinois.

2. Allied General Nuclear Services Facility, Barnwell, South Carolina.

3. Nuclear Fuel Services, Inc., West Valley, New York.

Also in late 1977, the Department commenced a comprehensive examination of the prospective needs of U.S. utilities for AFR storage. Since that time, a dialogue has been continuing between the Department and the utilities with respect to AFR storage needs and the schedule therefor. The Department has encouraged the utility companies to maximize the use of spent-fuel storage facilities at nuclear power plants in order to minimize the amount of AFR storage capacity that will be needed.

V.A STORAGE CAPACITY REQUIREMENTS

In developing a plan providing for prospective AFR storage needs, the Department has made a series of determinations regarding the amount of spent fuel that would be discharged from U.S. nuclear power reactors in excess of that which could be stored on-site at individual reactor storage facilities. The Department's program for determining AFR spent-fuel storage requirements focused first on obtaining the necessary data from the utility industry, and second on the preparation of a computer model to assist in analyzing the data. The information obtained from the utilities included the following:

1. The number of reactors presently operating, under construction, or committed to be constructed, along with various operating assumptions, such as plant factor and fuel burnup, from which a total spent-fuel discharge schedule could be computed.

2. Present and planned capacities of spent-fuel storage basins at reactors, together with an analysis of capabilities and plans for expansion of this storage capacity.

3. The utility plans for transshipment* of spent fuel between reactors and the utility philosophy with respect to maintaining reserve storage capacity for an entire fuel core.**

The Department's analysis of AFR storage requirements also has taken into consideration the quantity of foreign fuel that might be delivered to the United States for storage under the President's October 1977 spent fuel policy.

In February 1979, the Department published a report entitled "Spent Fuel Storage Requirements--The Need for Away-From-Reactor Storage" (4). In June 1979, the Department received an update of information from the utilities and published the results thereof in February 1980 (5).

Using the aforementioned information as a base, the following paragraphs describe the projections of nuclear power growth and the amount of spent fuel expected to result therefrom, the requirements for AFR storage of spent fuel, and the sensitivity of such requirements to changes in the projected growth of nuclear power. The sensitivity of AFR storage requirements to the disposal issues are discussed in Part VI.

V.A.1 Nuclear Power Growth and Spent-Fuel Discharges

Table V-1 shows a forecast of nuclear power growth through 2010 and an estimate of the cumulative quantity of spent fuel which will result therefrom.

---

*Transshipment consists of shipping spent fuel from one reactor of a utility company to the site of another reactor of the same type owned by the same company for the purpose of storage at the second reactor site.

**It is a desirable operating practice to reserve space in a reactor's spent fuel storage pool to permit at any time the discharge of all the fuel contained in the reactor into the pool so that the core internals can be inspected or repaired, if necessary.

Table V-1. Nuclear Power Growth Projections and Spent Fuel Discharges for the Period 1980-2010

Year	Nuclear Capacity (GWe)	Cumulative Discharges (1000s MTU)
1980	56	7.5
1981	61	9.1
1982	76	10.9
1983	92	13.1
1984	104	15.7
1985	125	18.7
1986	136	22.3
1987	148	26.2
1988	155	30.4
1989	160	34.8
1990	171	39.4
1991	186	44.1
1992	197	49.0
1993	207	53.9
1994	213	60.5
1995	224	66.3
1996	234	72.5
1997	245	78.8
1998	255	85.5
1999	266	92.5
2000	276	99.7
2001	294	107.2
2002	312	115.0
2003	330	123.3
2004	348	132.2
2005	366	141.4
2006	384	151.1
2007	402	161.3
2008	420	171.8
2009	438	182.9
2010	456	194.4

In Table V-1, the projection of nuclear growth through 1994 is based on reactors currently operating, under construction, or committed. This projection is consistent with the high growth projection in the Energy Information Administration (EIA) Annual Report to Congress (6).

The projection of nuclear growth from 1995-2010 is based on annual additions of 10.5 GWe for 1995-2000 and 18.0 GWe for 2001-2010. These values are based on extrapolations from the EIA report.

Most of the spent fuel discharged as shown in Table V-1 will be stored in reactor storage pools. The AFR storage requirements associated with these discharges are discussed in V.A.2. Spent fuel discharges and corresponding AFR storage requirements for a lower nuclear growth rate are discussed in V.A.3.

V.A.2 Away-From-Reactor Storage Requirements

The requirements for AFR storage capacity are dependent on utility plans for on-site storage capacity at reactors, the extent of transshipment of spent fuel between reactor basins, and utility philosophy for reserving a full core discharge capacity in the reactor storage pools. The Department's staff has examined the prospective ranges of storage requirements resulting from variations in these factors and has developed several scenarios for AFR storage facility planning. These cases represent a series of projections of AFR storage requirements which cover the expected range of prospective AFR needs in the future. Table V-2 lists six cases of interest, and Figure V-1 shows the corresponding AFR storage requirements.

Table V-2. Cases of Interest

CASE I	- Licensed Expansion Plans No Transshipment
CASE II	- Licensed Expansion Plans Intra-utility Transshipment
CASE III	- Current Expansion Plans No Transshipment
CASE IV	- Current Expansion Plans Intra-utility Transshipment
CASE V	- Maximum Expansion No Transshipment
CASE VI	- Maximum Expansion Intra-utility Transshipment

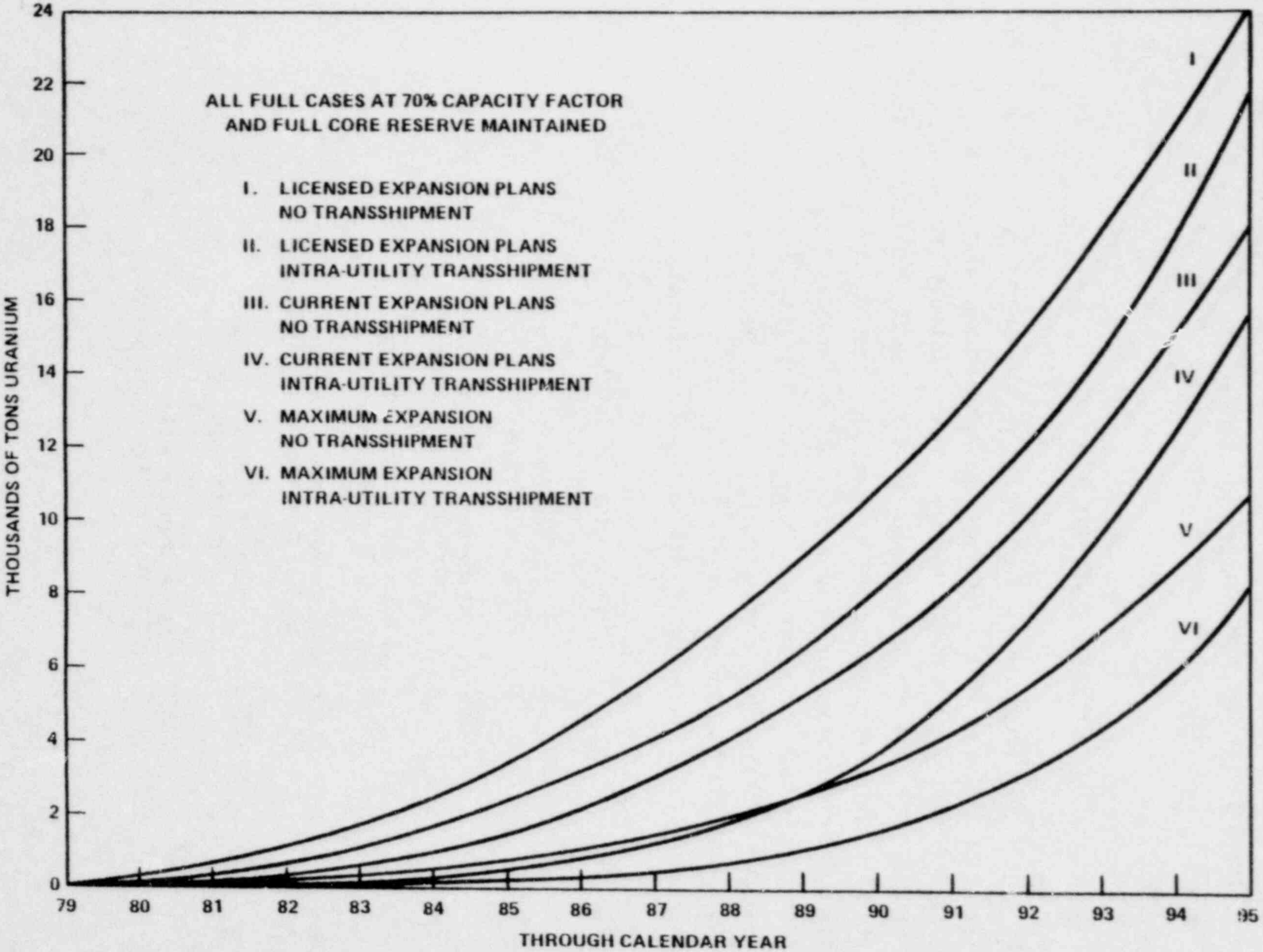


Figure V-1. Comparison of Cumulative Away-from-Reactor Storage Needs
Cases of Interest

The highest AFR storage requirement results from the assumption that there will be no further expansion of reactor storage pool capacity beyond that currently licensed and that no transshipment of fuel between reactors will take place (Case I). Such a situation would result if there were a restriction on fuel storage at reactors and on shipping between reactors. Conversely, the lowest AFR storage requirement results from the assumption of transshipping and the maximum expansion of reactor storage pools (Case VI).

Of the six cases of interest in Table V-2, three are of particular significance to the Department spent fuel storage program:

1. Case III is based on the utilities expanding their storage pools according to their current plans; it does not assume transshipment between reactors, unless transshipment is actually occurring. This case represents a reasonable upper bound for planning purposes.

2. Case V assumes that the utilities expand their reactor storage pools to the maximum capability they have estimated. As in Case III above, transshipment is not assumed unless it is actually occurring. Recent information from the utilities confirms that this case best represents the situation likely to occur. Most utilities are moving toward maximum expansion of their existing reactor storage pools; however, many have indicated that transshipment should not be included in the base case planning. Hence, the maximum expansion-no transshipment case has been chosen as the basis for planning purposes.

3. Case VI involves the same expansion plans as Case V (i.e., maximum expansion), but assumes transshipment of fuel between the same types of reactors (PWR or BWR) belonging to the same utility. This case is used as a lower bound for planning purposes.

These three cases are shown graphically in Figure V-2.

Upper Bound --Current expansion plans, no transshipment

Base Case --Maximum expansion, no transshipment

Lower Bound --Maximum expansion, intra-utility transshipment

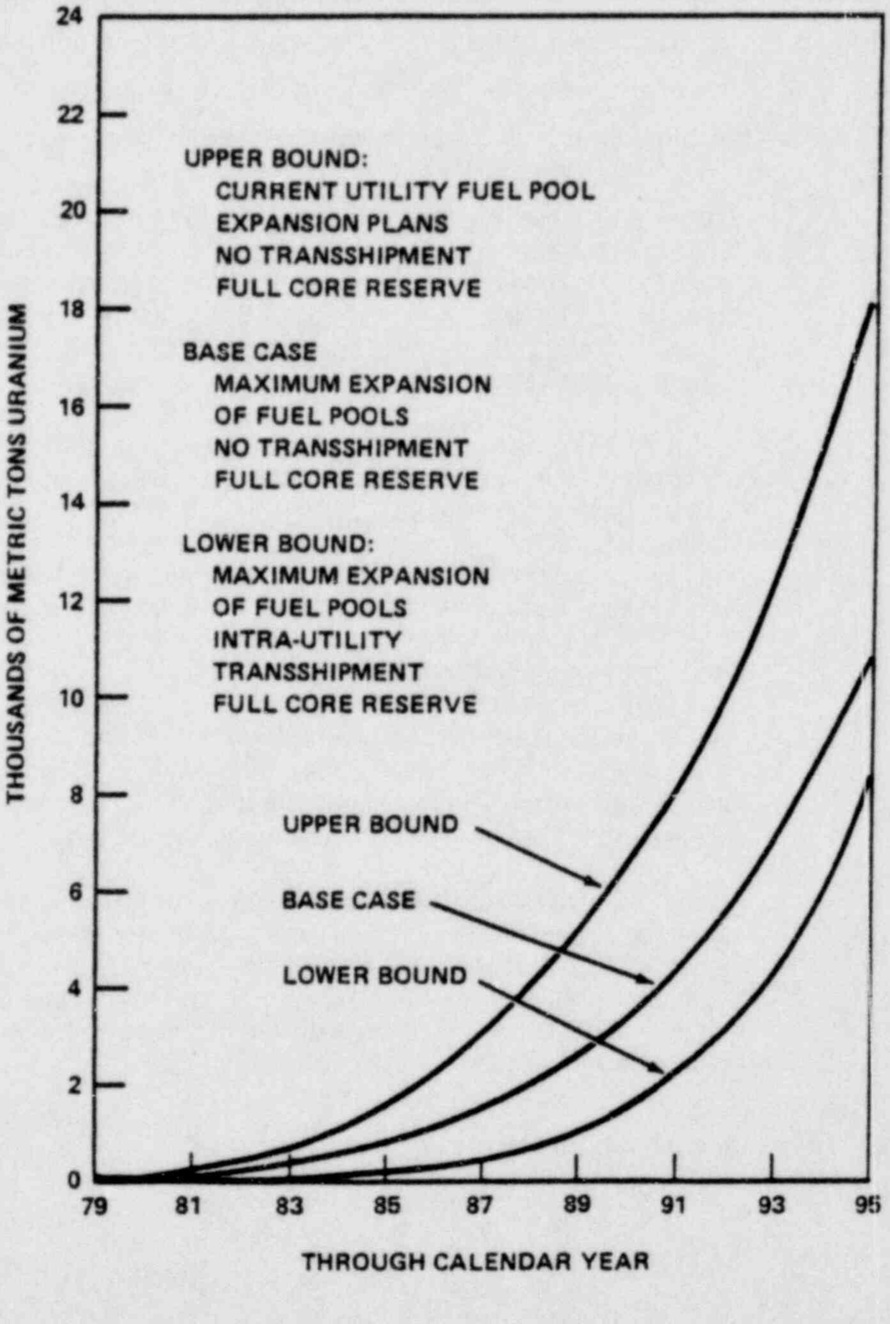


Figure V-2. Range for Away-from-Reactor Requirements for Planning Purposes

In view of both the trend of utilities to expand the capacity of their reactor pools as much as possible through reracking and the uncertainty of the extent to which intra-utility shipments could be utilized, Case V (maximum expansion with no transshipment) is a prudent basis for planning purposes. Table V-3 shows the cumulative AFR storage requirements for this case through 2010, based on the nuclear growth projection set forth ir Table V-1, assuming that no repository is available by that time.

Table V-3. Cumulative Quantity of Spent Fuel in Excess of At-Reactor Storage Capacity

Year	Maximum Utility Expansion Plans No Transshipment Domestic Fuel
1983	0.4
1984	0.5
1985	0.8
1986	1.0
1987	1.5
1988	2.0
1989	2.5
1990	3.3
1991	4.3
1992	5.5
1993	7.0
1994	8.8
1995	10.9
1996	13.3
1997	16.1
1998	19.2
1999	22.7
2000	26.3
2001	30.3
2002	34.5
2003	38.5
2004	44.8
2005	49.5
2006	55.1
2007	60.5
2008	67.8
2009	72.5
2010	79.0

Figure V-3 shows the cumulative projected amounts of spent fuel (excluding foreign fuel) stored at reactor sites and at AFR storage facilities during the period 1981-1995. From the information set forth in this figure, it can be seen that most of the spent fuel discharged from commercial nuclear power plants will be stored by the utilities in reactor storage facilities, with only a small percentage being stored at AFR storage facilities.

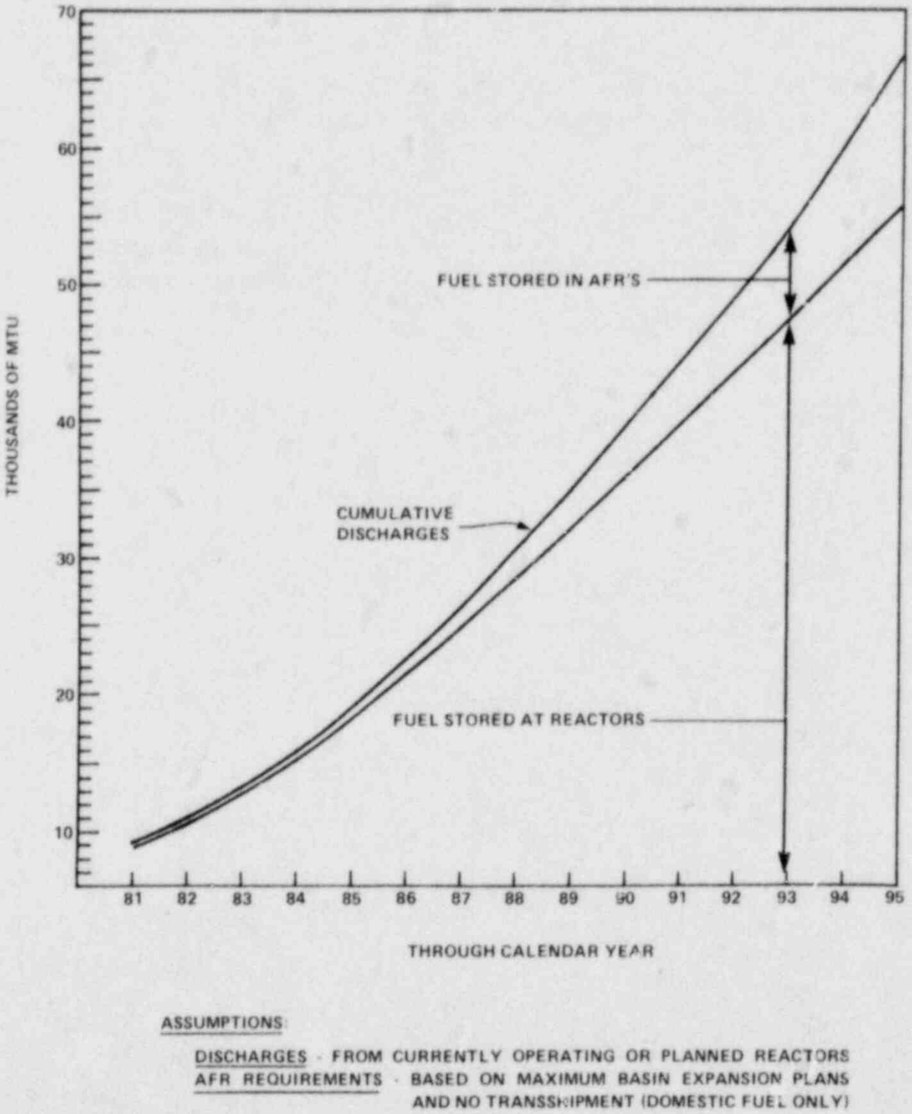


Figure V-3. Projected Requirements for Storage of Spent Fuel at Reactors and Away from Reactors for the Period 1981-1995 for the Assumed Case (Case V)

Source: (Reference 5) U.S. Department of Energy, Spent Fuel Storage Requirements - The Need for Away-From-Reactor Storage, DOE/NE-0002, February 1980

V.A.3 Sensitivity of Storage Requirements to Nuclear Power Growth Projections

The impact of a change in nuclear power projections on storage requirements has been studied by the Department. A sensitivity analysis was conducted in which a lower growth of nuclear electric generating capacity was assumed--186 GWe of installed nuclear capacity in the year 1995 (as opposed to the 224-GWe capacity figure in Table V-1). This is consistent with the low projection from the EIA Annual Report to Congress. The estimated quantities of spent fuel to be discharged from reactors for this case are shown in Table V-4.

Table V-4. Lowest Growth Projection Case for Nuclear Power Capacity

Year	Nuclear Capacity (GWe)	Cumulative Discharges (1000's MTU)
1980	56	7.5
1981	61	9.1
1982	76	10.9
1983	92	13.1
1984	104	15.7
1985	125	18.7
1986	136	22.3
1987	148	26.2
1988	155	30.4
1989	160	34.8
1990	168	39.2
1991	172	43.7
1992	175	48.3
1993	177	52.9
1994	178	57.5
1995	186	62.3
1996	196	67.4
1997	207	72.8
1998	217	78.4
1999	227	84.3
2000	238	90.5
2001	256	97.2
2002	274	104.3
2003	292	111.7
2004	310	120.0
2005	328	128.5
2006	346	138.4
2007	364	147.9
2008	382	157.8
2009	400	168.2
2010	418	179.1

The Department has analyzed actual utility data on maximum reactor basin reracking and found that the projected AFR requirements lag the spent fuel discharges by about 13 years (the average storage capability of reactor storage pools). However, it is possible that future reactors which can still change the design of their reactor storage pools will provide more storage than current generation reactors. Since the spent fuel which would be shipped to AFR

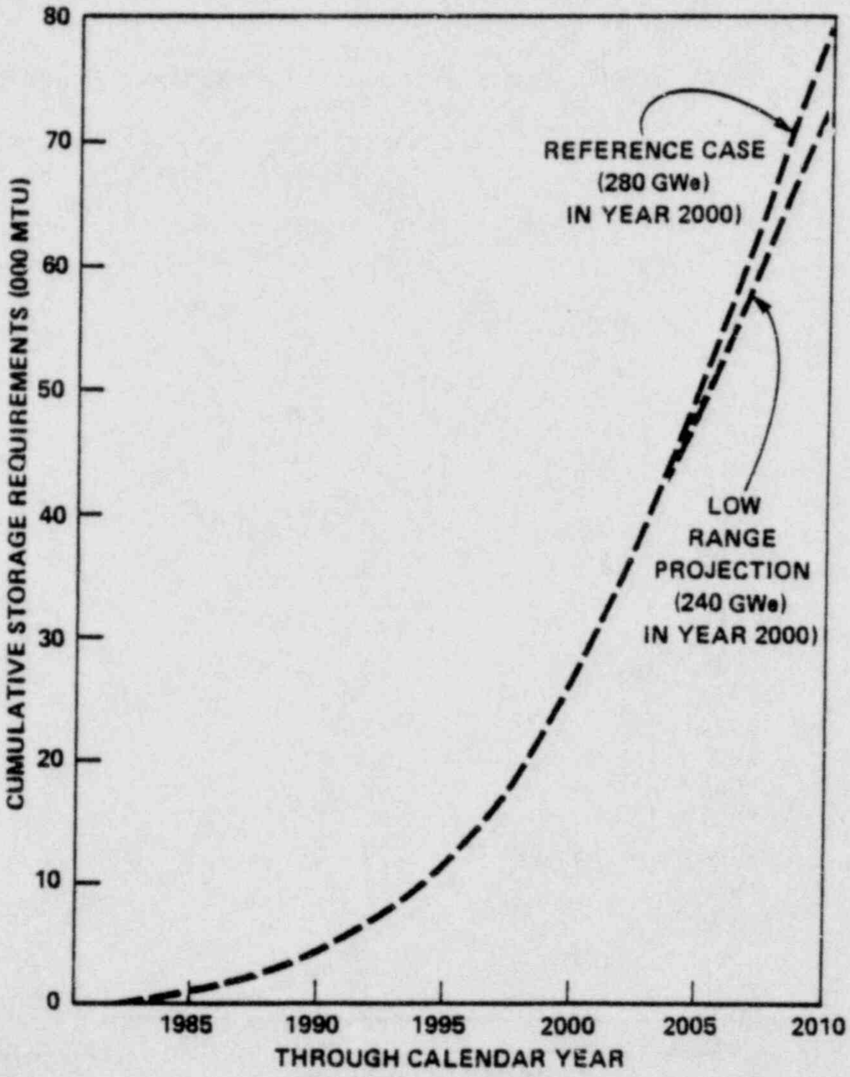


Figure V-4. Projected Away-From-Reactor Requirements as a Function of Nuclear Power Growth storage facilities in the near term is expected to be from reactors already built or committed, the impact of the change in the nuclear growth projection is not seen in the near term, as Figure V-4 illustrates. This means that the near-term AFR requirements and early repository demands are unaffected by moderate changes in the nuclear growth projections.

V.B INSTITUTIONAL CONSIDERATIONS

This chapter describes the institutional considerations involved in the development of the AFR storage capacities needed to meet the projected requirements for such storage, as described in the preceding chapter, and the steps that are being taken by the Department to resolve the institutional issues involved.

V.B.1 Regulation and Licensing

The principal regulatory decisions that must be made prior to the licensing of new AFR spent fuel storage facilities concern the finalization of NRC regulations and regulatory guides. NRC regulations and regulatory guides have been issued in draft form, comments have been received thereon, and they are now in the process of finalization (7, 8). A Final Generic Environmental Impact Statement has been issued by the Commission on spent fuel storage activities (9). Draft generic environmental impact statements on spent fuel storage have been issued by the Department of Energy (10-13), comments have been received thereon, and final statements will shortly be issued. Thus there do not appear to be any impediments to prompt promulgation of regulations affecting spent fuel storage or to prompt compliance with the NEPA requirements that are associated therewith.

Licensing of water basin storage of spent fuel has been routinely practiced in the United States both at nuclear power plant sites and reprocessing plant sites for nearly 20 years and is continuing (IV.D.1). The proposed new regulations and regulatory guides affecting AFR storage facilities are less rigorous than those currently applicable to reprocessing plants and nuclear power plants because of the relatively benign nature (9, 11) of the storage activity involved (i.e., storage only, in contrast to storage in combination with the operation of a reactor or a reprocessing facility). Using the existing knowledge which has been gained through the past 30 years of experience, AFR storage facilities can be engineered to meet the existing and proposed regulations and any reasonable modifications thereto, including those pertaining to the physical protection of facilities and the accounting and control of nuclear materials.

V.B.2 Cooperation with States

The Department recognizes the concern of the States and plans to follow a program of cooperation with States containing likely sites for new AFR storage facilities. This program will explain the nature of the facilities and the operations conducted therein and encourage States to give meaningful advice through consultation and participation in applicable proceedings.

Prior to initiating any action with respect to the siting of an AFR storage facility in a State, the Department plans to contact representatives of the governor of the affected State to solicit assistance and to determine the degree and type of participation desired. Preliminary meetings will be held with officials of affected States. The States will have the opportunity to participate in hearings with respect to the siting of AFR storage facilities. In addition to participation in hearings, other activities (such as environmental and engineering studies at specific sites) will be discussed with appropriate State officials. Cooperation with State officials will involve providing detailed technical information to them and giving careful consideration to advice from them.

V.B.3 Congressional Authorization

Both Congress and the Administration recognize the prospective need for AFR storage facilities. Administration-sponsored bills (14), introduced in March 1979, would authorize the Department to provide interim storage of spent fuel from domestic and foreign nuclear power plants, including financing of the necessary facilities involved. These bills are currently under consideration by the Congress.

V.B.4 Implementation of the National Environmental Policy Act

The National Environmental Policy Act of 1969 (NEPA) (15), as implemented by the regulations of the Council on Environmental Quality (CEQ) (16) and the Department's guidelines (17), requires that the potential environmental consequences of proposed actions be considered in the Department's planning and decisionmaking. The Department has developed a NEPA implementation plan for the AFR storage program which will ensure the integration of the NEPA process into overall Department planning and decision making. The Department's plan is based on the "tiered" approach, which is designed to eliminate repetitive discussion of the same issues and to focus on the actual issues ripe for decision at each level of environmental review. Under this approach, general matters are covered in broad programmatic environmental impact statements (EIS's) with subsequent narrower EIS's or environmental assessments (EA's) incorporating by reference the analysis of general issues and concentrating only on the issues specific to the subsequent decision.

Pursuant to its NEPA plan, the Department has prepared draft EIS's on the various elements of the President's policy for the storage of spent fuel. These draft EIS's, which were issued in December 1978 (11-13), analyze the generic environmental consequences of implementing a spent fuel storage program and alternatives to such a program. Under the proposed spent fuel storage program, the Department would take title to and store domestic spent fuel at one or more AFR storage facilities pending the availability of a repository for ultimate disposal or of a reprocessing plant if a decision is made in favor of reprocessing in the future.

If a decision is made to implement the proposed spent fuel storage program, the Department plans to prepare an AFR spent-fuel storage facility EIS which will provide environmental input into decisions on the selection of facilities to meet the projected near-term demand for AFR storage capacity. This EIS will analyze the need for the Department AFR capacity, using the latest available data concerning utility plans for expansion of on-site capacity, compaction, transshipments, and the expected quantities of spent fuel discharges. As planned, this EIS will focus on options for meeting the near-term AFR storage requirements. Such options are limited to the use of existing commercial facilities, modified as necessary by reracking but with no new construction.

The site-specific environmental impacts, including transportation impacts, of potential use of three existing commercial sites as Department AFR's will be analyzed. These sites are the General Electric Facility at Morris, Illinois (GE-Morris); the Allied General Nuclear Services Facility at Barnwell, South Carolina (AGNS-Barnwell); and the Nuclear Fuel Services, Inc., Facility at West Valley, New York (NSF-West Valley).

In addition, the EIS will analyze the impact of not providing AFR capacity to satisfy the projected near-term AFR storage requirements, but rather delaying implementation of the program until new construction can be completed. Department decisions for which this EIS will provide environmental input are: (i) acquisition of one or more of the existing commercial facilities, (ii) adaptation of the facilities through reracking, and (iii) submission of licensing applications to NRC to operate the facilities as AFR facilities. In addition, the EIS will provide input to programmatic decisions on which options to pursue to meet long-term AFR capacity requirements.

Subsequent EIS(s) will analyze the environmental impacts of adding new facilities at existing sites and constructing new AFR facilities at specific new sites to meet the long-term demand. This EIS will serve as input for decisions on acquisitions of sites, construction of facilities, and license applications for the options analyzed.

The status of the repository schedule and spent-fuel storage requirements will be reviewed continually to ensure that as additional AFR storage requirements are projected, appropriate NEPA analysis is begun to serve as input for decisions on options to meet these additional requirements.

In conclusion, the Department's AFR management plan is structured to integrate NEPA requirements into overall program planning and decision making. Meaningful consideration of the environmental impact of all reasonable alternatives at each stage of the decisionmaking process will be assured. The Department believes that this management approach will result in the timely attainment of a safe and environmentally acceptable AFR program.

V.C IMPLEMENTATION OF AWAY-FROM-REACTOR PROJECTS

This chapter describes the steps that the Department is taking to implement development of a government AFR spent-fuel storage capability, the availability of facilities for AFR storage activities, and the prospective schedule for developing of AFR storage capability.

V.C.1 Department of Energy Organization

As noted in III.B.1, the President has given the Secretary of Energy overall responsibility for integrating the Nation's nuclear waste management program. Within the Department of Energy, day-to-day responsibility for the nuclear waste management program is assigned to the Assistant Secretary for Nuclear Energy, who reports to the Undersecretary and the Secretary. The Deputy Assistant Secretary for Nuclear Waste Management directs the Office of Nuclear Waste Management (ONWM) and is responsible for executing policy and managing all aspects of the nuclear waste management program for both disposal (the NWTS Program) and Storage. (See III.B and Figure III-1.) Within ONWM, the Director of the Division of Transportation and Fuel Storage has overall responsibility for the Department's AFR spent-fuel storage program and provides guidance to the Spent Fuel Project Office on the policy to be followed in the implementation of project activities. The Department established (in November 1978) the Spent Fuel Project Office at its Savannah River Operations Office, which has been assigned the prime responsibility for spent fuel project management activities, including planning and execution functions.

The Department's AFR spent fuel storage program is directed toward providing the capacity to store the spent nuclear fuel discharged from commercial power reactors in excess of that which can be stored at reactor storage facilities. The program is implementing the policy to accept and take title to spent nuclear fuel from the domestic utilities and to store limited amounts of foreign fuel when such action would contribute to meeting nonproliferation goals. Management and implementation of government activities related to the storage of spent fuel include the following:

1. Managing facilities, operations, and resources.

2. Providing storage capacity.

3. Evaluating and influencing interfacing systems (utilities, transportation, repository).

4. Developing advanced fuel storage technology.

5. Managing the foreign fuel storage program.

The Spent Fuel Project Office has the responsibility to:

1. Identify and predict the needs for storage capacity, including evaluation of predictions by others such as the Nuclear Regulatory Commission, utility groups, and independent contractors. Establish and maintain a current assessment of foreign spent fuel storage needs, capabilities, and plans.

2. Explore all options for spent-fuel storage, (both existing options and future possibilities) which requires developing information needed to make decisions on which options to pursue.

3. Evaluate and plan modifications to existing facilities and their operating techniques.

4. Identify and analyze space requirements for the storage of spent fuel.

5. Develop and qualify technology related to spent fuel storage to fill gaps in existing technology.

6. Perform site evaluation studies to aid in site selection and qualification to maintain construction of an AFR storage facility as a viable option for spent-fuel storage.

7. Review and recommend to the Director, Division of Transportation and Fuel Storage, approval of selection of architect engineers, construction managers, and operating contractors for AFR storage facilities.

8. Ensure compatibility of AFR and geologic facilities for storage and disposal of spent fuel.

9. Identify, anticipate, and cause to be resolved safety and environmental issues related to spent-fuel storage.

10. Evaluate the needs and capabilities of the utilities and carriers to ship spent fuel.

11. Interact with the radioactive materials transportation industry, carriers, utilities, Nuclear Regulatory Commission and other government agencies, both Federal and State, to coordinate licensing, transportation, and other activities. Transportation activities will be coordinated with the Department's Albuquerque Operations Office and Sandia National Laboratories. (See Part VI.)

12. Establish requirements for a quality assurance (QA) program and ensure proper implementation of the QA program by all prime contractors. Ensure that the QA program is conducted in a format acceptable to the Nuclear Regulatory Commission.

13. Maintain technical overview of project activities by means such as periodic meetings, reviews, and project reports, to ensure that AFR storage project objectives are being met effectively.

14. Perform project administrative functions including planning and scheduling, contracting, contract administration, budget administration, and cost control.

15. Establish and operate a business office to identify and support actions to obtain any new legislative authority for contracting with utilities for the storage services and to collect and revise fees. The business office will identify requirements for contracting for storage space in private facilities as appropriate. In addition, the business office will collect fees, contract with utilities, operate facilities as appropriate, contract for decontamination and decommissioning, and ship spent fuel to a geologic repository. The office also will support the Department in the contracting for storage space in private facilities.

Thus there is in place a management program, designed to provide and maintain an AFR spent-fuel storage capability once the necessary funds are appropriated for full implementation of the Department's spent fuel storage program.

V.C.2 Selection of Candidate Sites for Storage

There are three existing facilities that could be used in their present or expanded form for AFR storage of spent fuel. These are the reprocessing facilities of GE-Morris; Nuclear Fuel Services, Inc. NFS - West Valley; and AGNS - Barnwell. Both the GE and NFS facilities currently are licensed to store spent fuel and in fact now are storing spent fuel. Although the AGNS facility is not yet licensed to receive fuel, an environmental and licensing review has been performed on it. It appears that these three existing sites would be suitable for spent fuel storage.

With respect to new AFR storage facilities, all the site selection criteria currently exist for such facilities (7, 18). Because of the relatively benign nature of AFR storage facilities, many locations in the United States would meet these technical criteria when the facilities are specifically designed for a particular location.

V.C.3 Storage Facility Availability

The provision of AFR spent-fuel storage capacity in the near term (8-9 years) will require the use of existing spent-fuel storage capacity. Only the existing facilities at GE-Morris, AGNS-Barnwell, or NFS-West Valley could be available to provide near-term AFR storage. The existing available capacities of these facilities are approximately 350 MTU, 400 MTU, and 90 MTU, respectively (19). The GE and AGNS facilities alone could provide sufficient capacity to furnish the projected needs for storage capacity through 1984. Barnwell could be reracked to provide a total of 1,750 MTU capacity in 1984. If required, new storage facilities at existing sites or at new sites could be brought on at a later date to meet long-term requirements.

Longer term requirements would necessitate construction of new AFR facilities or the addition of new pools at existing sites; the Department estimates that a new facility could be designed, licensed, constructed, and operating in 83 months at an existing nuclear site or 95 to 98 months at a new stand-alone site, as follows:

Step	New Sites	Existing Sites
Site selection	12 months	
License preparation	15-18 months	15 months
NRC review	32 months	32 months
Construction and startup	36 months	36 months
Total	95-98 months	83 months

With these storage facility options, it is clear that firm plans can be fashioned to provide limited storage capacity for the near term and to meet future storage requirements as needed. The use of existing storage facilities depends upon completion of negotiations of appropriate financial settlements with their owners. Existing facilities could be put to beneficial use, small increments of capacity could be provided as needed, and a growth potential would exist. Table V-5 sets forth a summary of the capacities of these facilities, the time required for expansion thereof, and the estimated capital cost of expansion of the facilities.

The Department also is conducting research and development programs on methods of further increasing storage capacities by such means as rod storage (20, 21), sealed cask storage (22), and dry vault storage at GE-Morris and AGNS. Such techniques might well be incorporated into both at-reactor facilities and AFR facilities in the future to provide additional storage capacity for spent fuel as well as program flexibility.

V.C.4 Overall Schedule for Away-From-Reactor Storage Facilities

The exact schedules for the startup of AFR storage facilities are dependent on which facilities are selected. As Table V-5 shows, there is a wide variation between the possible schedules for the different facilities.

Table V-5. Away-From-Reactor Facilities

Facility		Total Capacity (cumulative MTU)	Schedule (cumulative years)	Cost (cumulative $millions)
GE-Morris:	Existing	350	2-1/2	x
	Reracked	750	2-1/2	x+9
AGNS:	Existing	400	3-1/2	y
	Reracked	1,750-2,350	3-1/2	y+25
NFS:	Existing	90	+	+
	Reracked	1,500	+	+
New Facility:		5,000	8*	303**

x - negotiated acquisition cost of GE-Morris.
y - negotiated acquisition cost of AGNS.
* - including 1 year for site selection.
+ - not yet estimated.
**- 1980 dollars.

The ultimate selection of specific facilities will depend on the outcome of careful examination of the alternatives and on the negotiations for the acquisition of use of existing facilities. Figure V-5 shows the times required for bringing AFR storage capacity on line after Congressional authorization has been obtained.

Figure V-6 shows one possible approach to providing AFR storage capacity. It is assumed that the existing facilities are reracked and then a new AFR is built. Based on current projections, the new facility would be required around 1990. Site selection for the new facility would occur in 1982.

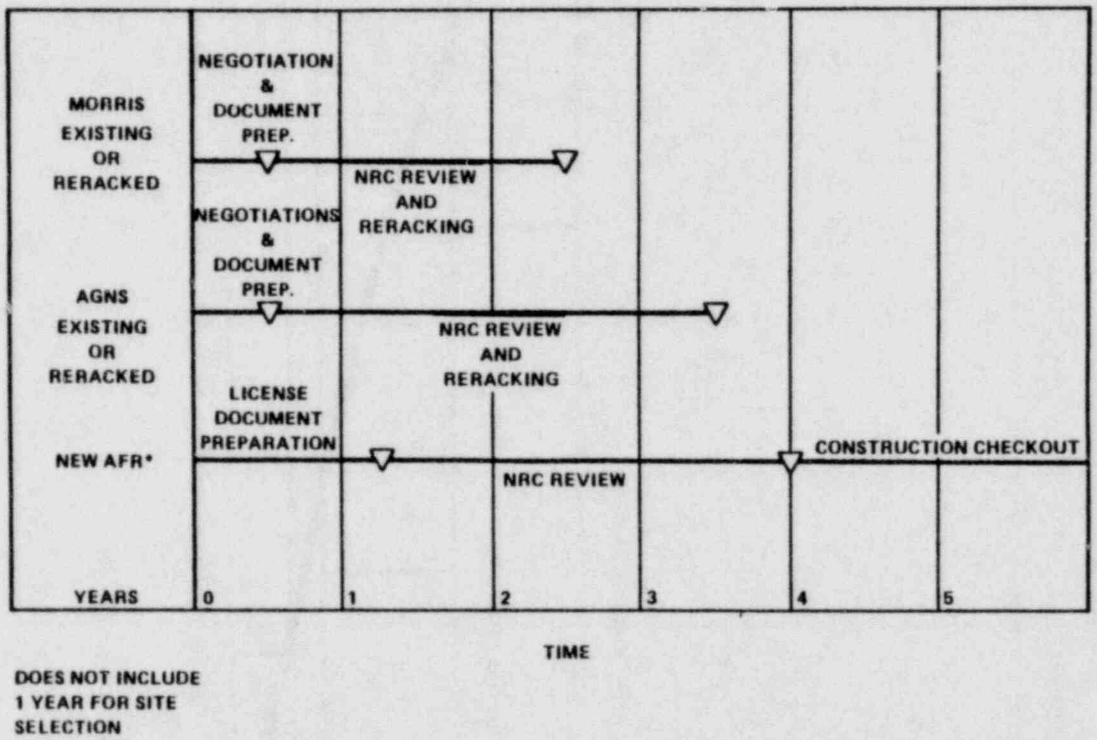


Figure V-5. Projected Away-From-Reactor Facility Availability Schedules

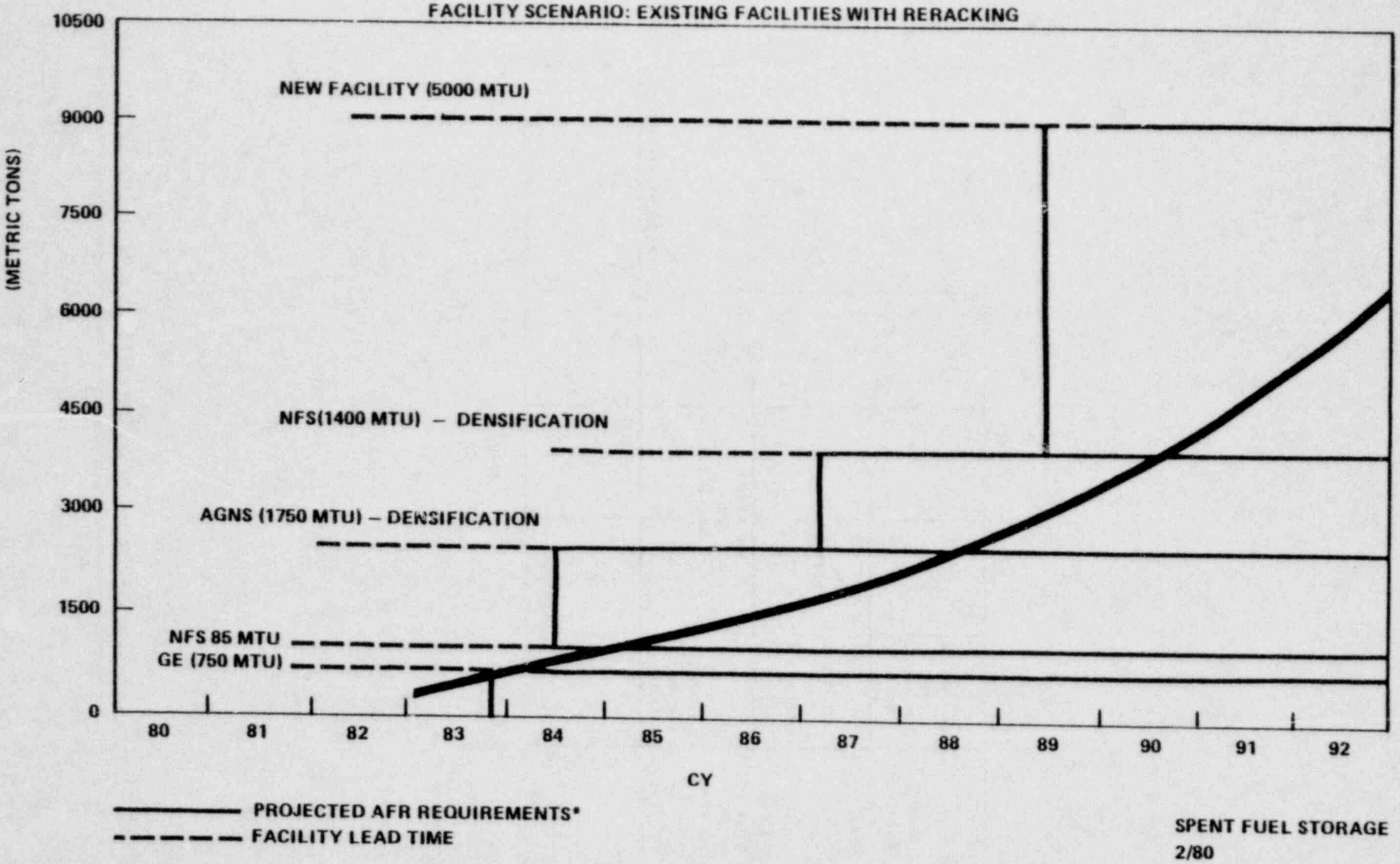


Figure V-6. Possible Facility Development to Meet Projected Spent-Fuel Storage Requirements

V.D COSTS OF AWAY-FROM-REACTOR STORAGE FACILITIES

The capital cost of a new AFR storage facility having capacity to store 5,000 MTU as spent fuel and a capability for receiving 1,000 MTU/yr has previously been estimated to amount to about $250 million in 1978 dollars, as shown in Table V-6 (23).

Table V-6. Capital Cost Breakdown For 5,000 MTU Away-From-Reactor Spent Fuel Storage Facility ($millions)

Item	Amount
Storage pool and building	$ 65.6
Equipment for storage pool and building	59.9
Auxiliary and administrative buildings and equipment	47.0
Engineering and design	31.3
Contingency	46.2
Total	$250.0

In addition to the capital costs set forth above, it is estimated that the annual costs for operation will amount to $6 million for years in which spent fuel is being received or shipped by the facility, and will amount to $4 million for years in which only storage functions are being performed (1978 dollars). Escalation to 1980 dollars would increase these figures by approximately 21%.

Decommissioning costs for the AFR have been estimated to amount to about 10% of the initial capital cost of the facility, or $25 million (1978 dollars).

V.E SUMMARY OF MANAGEMENT PROGRAMS FOR PROVIDING AWAY-FROM-REACTOR STORAGE FACILITIES

The Department has estimated that AFR spent fuel storage facilities can be made available commencing as early as 3-4 years after Congressional authorization and that the necessary AFR storage capacity can be maintained until geologic repositories are available for the disposal of stored fuel. This estimate is based on the information summarized in V.A through V.C, which demonstrates the following:

1. The near-term (through 1990) needs for AFR storage capacity can be satisfied by acquisition and expansion of the storage capacity of existing facilities. The longer term needs for storage can be satisfied by building additional AFR storage facilities to supplement existing and expanded facilities. Needed capacities could be made available in existing storage facilities approximately 3 years after Congressional authorization and could be made available in new AFR storage facilities within 95 months after such authorization.

2. The Department has established an organization for the development and operation of AFR storage facilities.

3. There is sufficient information available on which to base selection of new sites for AFR storage, and the Department has embarked on a program designed to seek State involvement in the selection of proposed sites.

4. Finalization of new regulations pertaining to AFR storage of spent fuel is under way, and the technical ability to meet such licensing requirements exists.

5. Legislation has been submitted to Congress for authorization to acquire the necessary AFR spent fuel storage capability.

6. Analysis of the requirements for AFR storage capacity in the near-term and of the steps which must be taken to comply with the National Environmental Policy Act and other licensing requirements indicates that the necessary capacity can be provided on a timely basis.

References for Part V

(1) U.S. Department of Energy, "DOE Announces New Spent Nuclear Fuel Policy," News Release, October 28, 1977

(2) Presidential Message to Congress, 12 February 1980, "Comprehensive Radioactive Waste Management Program," Weekly Compilation of Presidential Documents, Vol. 16, No. 7

(3) R.S. Bell, W.N. Mobley, and J.M. Norris, Spent Fuel Program Preliminary Technical Assessment of Existing Facilities for AFR Storage Capability, DOE/SR/10007-1, U.S. Department of Energy, August 1979

(4) U.S. Department of Energy, Spent Fuel Storage Requirements - The Need for Away-From-Reactor Storage, DOE/ET-0075, February 1979

(5) U.S. Department of Energy, Spent Fuel Storage Requirements - The Need for Away-From-Reactor Storage, DOE/NE-0002, February 1980

(6) U.S. Department of Energy, Energy Information Administration, Annual Report to Congress, 1978, Vol. 3, pp. 203-231, DOE/EIA-0173/3

(7) 10 CFR, Chapter I, Proposed Part 72, "Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," (published October 6, 1978)

(8) U.S. Nuclear Regulatory Commission, Regulatory Guide 3.44, "Standard Format and Content for the Safety Analysis Report to be Included in a License Application for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (Water Basin Type)," December 1978

(9) U.S. Nuclear Regulatory Commission, Final Generic Environmental Impact Statement - Handling and Storage of Spent LWR Fuel, NUREG-0575, August 1979

(10) U.S. Department of Energy, Draft Environmental Impact Statement - Storage of U.S. Spent Power Reactor Fuel, DOE/EIS-0015-D, August 1978

(11) U.S. Department of Energy, Draft Environmental Impact Statement Supplement - Storage of U.S. Spent Power Reactor Fuel, DOE/EIS-0015-DS, December 1978

(12) U.S. Department of Energy, Draft Environmental Impact Statement - Charge of Spent Fuel Storage, DOE/EIS-0041-D, December 1978

(13) U.S. Department of Energy, Draft Environmental Impact Statement - Storage of Foreign Spent Power Reactor Fuel, DOE/EIS-0040-D, December 1978

(14) U.S. Congress, "Proposed Spent Nuclear Fuel Act of 1979," HR.2586 (Staggers-3/1/79) and S.797 (Jackson-3/27/79)

(15) National Environmental Policy Act of 1969, 42 U.S.C. 4321 et seq.

(16) Council on Environmental Quality - National Environmental Policy Act - Implementation of Procedural Provisions, Final Regulations, 43 Fed. Reg. 55978 (November 29, 1978) to be codified at 40 CFR Part 1500 et seq.

(17) Proposed DOE Guidelines for Compliance with the National Environmental Policy Act (44 Fed. Reg. 42135 (July 18, 1979)

(18) American Nuclear Society/American National Standards Institute, "Guidelines for Evaluating Site-Related Parameters for an Independent Spent Fuel Storage Installation," Draft Standard NXXX, September 1978

(19) Michael J. Lawrence, The Department of Energy's Spent Fuel Storage Program, AIF Workshop on the Management of Spent Fuel and Radioactive Waste, September 1979

(20) Allied General Nuclear Services, Spent Fuel Disassembly and Canning Programs at the Barnwell Nuclear Fuel Plant (BNFP), AGNS-35900-CONF-41, Barnwell, SC, October 1979

(21) Nuclear Assurance Corporation, Alternatives for Water Basin Spent Fuel Storage Using Pin Storage, SRO-1051-3 and NAC-C-7917 (Part III), Atlanta, GA, September 1979

(22) K.H. Henry and D.A. Turner, Storage of Spent Unreprocessed Fuel (SURF), CONF-78 0316-5, Rockwell Hanford Operations, Richland, WA, March 1978

(23) U.S. Department of Energy, Spent LWR Fuel Storage Cost, Reracking, AR Basins, and AFR Basins, DOE/SR-SF-2002, Rev. 1, January 1980

VI INTEGRATED OPERATION OF THE STORAGE AND DISPOSAL SYSTEMS

This part demonstrates the way in which integrated operation of the Department's overall waste management program will accommodate the continuing production of spent nuclear fuel. Management of the storage and disposal of spent fuel requires an integrated system which includes repositories, storage facilities, packaging facilities, and transportation networks. The elements of this system are highly interactive, i.e., repository availability, capacity, receiving rate, and deployment rate all have direct impacts on storage requirements. The relative locations of spent-fuel storage facilities and repositories influence transportation system considerations, including the number of casks required, shipment distances, and so forth.

Current Department program effort consists of the development of a total waste management system optimized in terms of costs and benefits. Specifc optimization studies will be performed in the near future. For illustrative purposes, assumptions can be made about the size, availability, and capacities of repository facilities and storage facilities. These assumptions allow the demonstration of the methods that are being used to develop an integrated spent-fuel management system and the assessment of the capability of the system to meet needs for the timely storage, transportation, and disposal of spent fuel.

The discussion that follows includes a reasonable scenario of repository capacity and deployment, the away-from-reactor (AFR) storage requirements resulting from this scenario, the sensitivity of AFR storage requirements to changes in the schedule for repository deployment, the transportation requirements associated with AFR storage and repository operations, the capabilities for meeting these requirements, and a general assessment of the overall cost of the waste management system.

VI.A AVAILABILITY RANGE FOR DISPOSAL FACILITIES

Part III of this Statement demonstrates that the first geologic repository for spent fuel will be ready for operation between 1997 and 2006. Subsequent repositories can be brought on line at 3-year intervals to meet spent-fuel disposal requirements. For the purposes of this analysis, it is assumed that the initial repository will have a total capacity of about 41,000 metrictons of uranium (MTU) of spent fuel, whereas subsequent repositories are assumed to have total capacities of about 69,000 MTU each. The assumption of a lower capacity for the initial repository corresponds to an assumption that the first repository will employ conservative loading conditions at a small site.

Preliminary design studies have shown that repositories could be able to receive and emplace spent fuel at rates of 1800 MTU/yr during their first 5 years of operation and 6,000 MTU/yr thereafter until they are filled to capacity (1). These studies addressed national repositories, i.e., facilities capable of receiving spent fuel from all the nation's reactors. Independent review of these studies (2) has confirmed the validity of designs which permit loadings at these rates. Additional studies (3) have addressed the maximum receiving rate allowable within the current scope of design.

In actual practice, there is considerable flexibility in determining the optimum receiving rate for repositories. In a system of multiple repositories, the spent fuel available for disposal can be allocated among several repositories. Thus, although the total spent fuel to be transported would remain constant, routes and cask requirements could be affected. It is anticipated that, even though an optimized system may not require receiving rates as high at any given repository as described here, individual repository designs will provide substantial margins in receiving ability to accommodate perturbations to the system, e.g., intentional shutdown of one repository.

VI.B AWAY-FROM-REACTOR STORAGE AVAILABILITY RANGE

As discussed in V.C.3, use of existing AFR storage facilities is planned to meet near-term needs. Existing facilities will provide sufficient additional storage capacity until the early 1990's. After this time, new AFR storage facilities can be constructed to meet AFR storage requirements until the geologic repository is available. Figure V-6 shows one possible option for satisfying near-term storage requirements, whereby initial requirements are met by acquisition and expansion of existing AFR storage facilities, followed by the construction of new AFR storage facilities.

VI.C AWAY-FROM-REACTOR STORAGE REQUIREMENTS

There will be a need for AFR storage of 13,300 MTU of domestic spent fuel through the year 1996 (see Table V-3). Assuming the first repository begins receiving spent fuel in July 1997, with subsequent repositories commencing operations at 3-year intervals thereafter, the maximum AFR storage capacity needed will be 20,000 MTU. Table VI-1 shows the predicted discharges of spent fuel, the cumulative AFR requirements, and the transfer of AFR-stored spent fuel to repositories for these assumptions of repository deployment.

As more analysis of total integration performance is done, tradeoffs will be made between the rate of development of repositories, the regional placement of repositories, the rate at which spent fuel might be placed in repositories, and the size and schedule of construction of AFR's. Accelerating the schedule for the second repository by 1 year would reduce the AFR capacity required by approximately 12%, whereas delaying the second repository for a 5-year interval would increase total AFR capacity required by about 14%. Analyses of this type will assist the optimum development of the total system.

VI.D SENSITIVITY OF AWAY-FROM-REACTOR STORAGE REQUIREMENTS TO CHANGES IN REPOSITORY SCHEDULING

Figure VI-1 shows the base-case demand for AFR storage and the amount of AFR storage needed if the initial repository startup occurs in mid-1997, in 2002, or in 2006. In each case, the initial repository receiving rate is expected to be 1,800 MTU/yr for the first 5 years; subsequently, the receiving rate is assumed to be 6,000 MTU/yr. Subsequent repositories start up at 3-year intervals. The maximum AFR storage requirements as a function of the startup date of the initial repository are as follows:

Initial Repository Startup Date	AFR requirements
July 1997	20,000 MTU
July 2002	44,000 MTU
July 2006	70,000 MTU

Table VI-1. Predicted Away-From-Reactor Storage Requirements (1,000 MTU)

(First repository operational in July 1997; additional repositories available at 3-year intervals)

Year	Transfers From Reactors (13 yr cooled fuel)	To Repositories	AFR Receipts (Discharges)	AFR Inventory
1996	-	-	-	13.3
1997	2.6	0.9	1.7	15.0
1998	3.0	1.8	1.2	16.2
1999	3.6	1.8	1.8	18.0
2000	3.9	2.7[a]	1.2	19.2
2001	4.2	3.6	0.6	19.8
2002	4.4	5.7	(1.3)	18.5
2003	4.6	8.7[b]	(4.1)	14.4
2004	4.7	9.6	(4.9)	9.5
2005	4.9	11.7	(6.8)	2.7
2006	4.9	7.6	(2.7)	-
2007	6.6	6.6	-	-
2008	5.8	5.8	-	-
2009	6.2	6.2	-	-
2010	6.3	6.3	-	-

[a]Second repository assumed to be in operation.
[b]Third repository assumed to be in operation.

The spent-fuel storage requirements shown in Figure VI-1 assume that fuel will be held in storage at reactor sites for 13 years after discharge. This time represents the average storage period for currently operating and planned reactors. New reactors are expected to have significantly greater storage capacity which may enable them to meet their storage requirements completely until a disposal facility is available. If all new reactors have provisions to meet their storage requirements the AFR requirements indicated in Figure VI-1 in the post-2000 time period would be reduced.

The actual pattern of repository loading will be based on safety, economic, and logistic considerations which prevail at the time repository loading begins. There are several alternate methods of repository loading which provide great flexibility in optimizing the operation of the total system. For example, it would be possible first to direct all spent fuel which

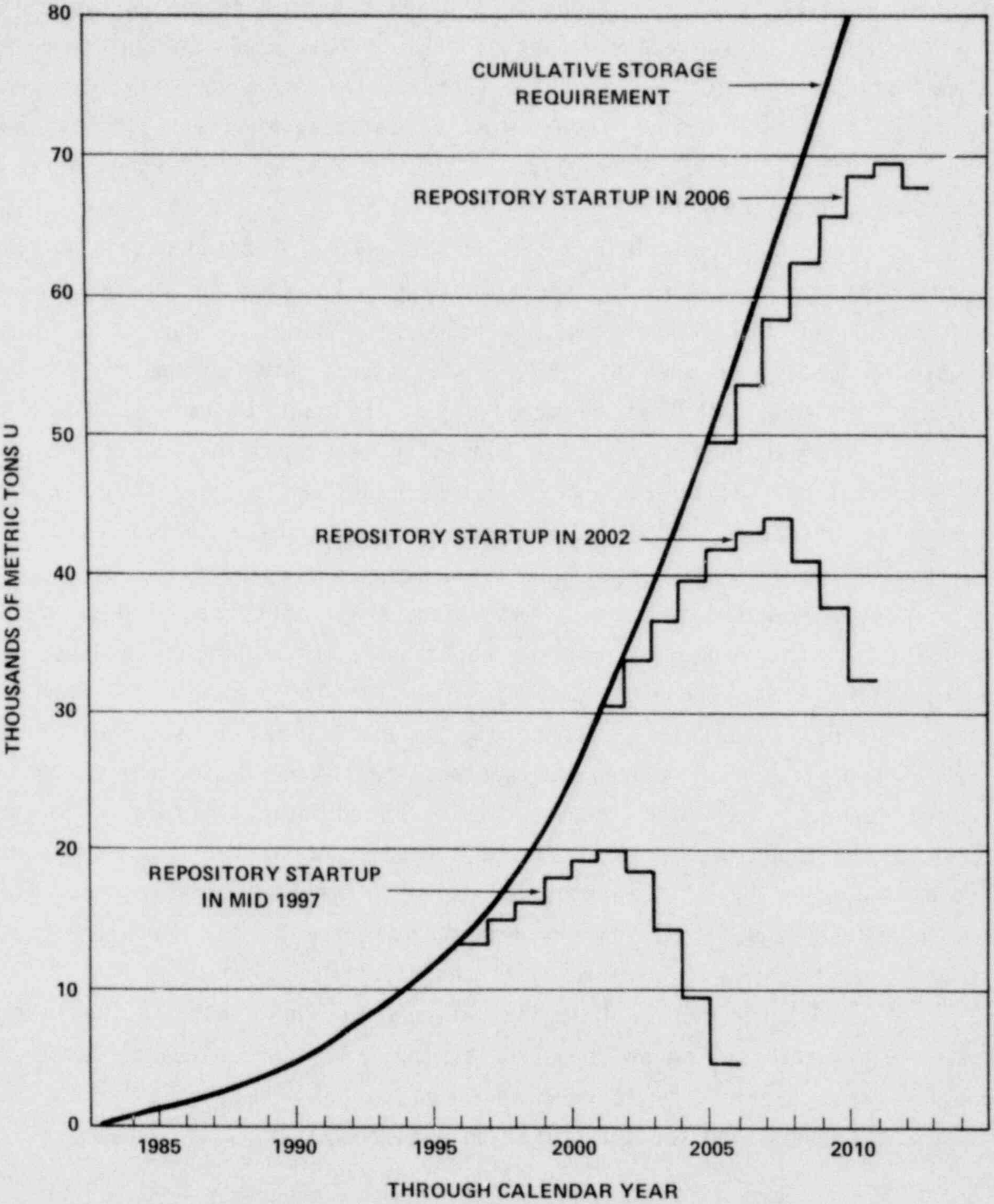


Figure VI-1. Away-From-Reactor Requirements as a Function of Repository Startup Dates had been in storage at reactor storage pools (the average residence of spent fuel in such pools is 13 years), up to the limit of the receiving capability of the repositories. The excess quantity of such fuel would be sent to an AFR storage facility. In any year that the repositories had a capacity to receive spent fuel in excess of that from domestic reactors, the repositories could receive spent fuel from AFR storage up to the capacity of the repositories to receive and emplace spent fuel.

A second possible method of repository loading might be first to transfer to the repository the spent fuel which had been in storage for the longest period of time. Approximately 5,000 MT of fuel is currently in reactor storage pools, so that by 1997, a significant initial loading for the repository would be fuel that has cooled for at least 17 years. This very conservative thermal loading could be placed in the repository during initial years in operation. Spent fuel which was being stored at reactors that were discontinuing operation would be transferred either to repositories or to AFR's from which the oldest fuel was being shipped, depending on the age of the fuel coming from the reactors. The disadvantage of the additional transportation that this might necessitate would have to be weighed against any advantages from this approach. In any year that the repositories had the capacity to receive fuel in excess of the amount of fuel in storage for the longest time or from fuel stored at reactors which were discontinuing operation, the repositories would receive fuel from domestic reactors up to the capacity of the repositories to receive and emplace spent fuel. In the event the receiving capacity of the repositories still had not been reached, fuel stored in AFR storage facilities would be transferred to the repositories to the extent that the receiving capacity of the repositories would permit.

Although detailed logistical analyses have not yet been completed, the combined system of repositories and AFR's have already been shown to provide great flexibility to meet the need to balance technical conservatism, regional needs, and reactor operation requirements.

VI.E TRANSPORTATION CONSIDERATIONS

Under government policy, the electric utility companies have the responsibility for transporting spent fuel from their reactors to either an AFR storage facility or a repository, whereas the government has the responsibility for transporting spent fuel from the AFR storage facility to the repository. It is not anticipated that transportation activities will limit the timing of disposal or AFR storage.

There are sufficient commercial organizations to provide shipping casks and services for the transport of spent fuel to AFR storage facilities or to repositories as needed and at competitive prices. As of January 1980, a total of 9 licensed truck casks and 6 licensed rail casks were available in the United States for transporting spent fuel. (Four truck casks are currently fabricated but not operational because the NRC has issued a show-cause order and their use is restricted until the owners respond. It is expected that some of these will be operational again in the near future.) A number of commercial organizations are involved in the design and construction of spent-fuel shipping casks; four of these companies offer a complete spent-fuel shipping service. In addition, there are other commercial organizations that have the capability of designing and constructing spent-fuel shipping casks (4).

The Department has established a Transportation Technology Center at Sandia National Laboratories to follow the emerging needs for spent-fuel transportation services and to establish contingency plans for providing the necessary shipping equipment (casks) and services in the unlikely event that commercial organizations are not able to meet the requirements on a timely basis. Guidance is provided to the Transportation Technology Center by the Departments Division of Transportation and Spent fuel (see V.C.1).

The following paragraphs describe the types of existing spent-fuel shipping casks owned by U.S. commercial organizations which currently provide spent-fuel shipping services:

1. NL Industries, Inc. (NLI) has two types of spent-fuel shipping casks, a legal weight truck system

(designated NLI 1/2) and a rail system (designated NLI 10/24). The NLI 1/2 cask is capable of transporting one PWR fuel assembly or two BWR fuel assemblies. The NLI 10/24 cask is capable of transporting 10 PWR fuel assemblies or 24 BWR fuel assemblies. There are five NLI 1/2 casks in operation at the present time, and two NLI 10/24 casks are in existence.

2. General Electric Co. has a rail shipping cask (designated IF-300), which is capable of transporting 7 PWR fuel assemblies or 18 BWR fuel assemblies. There are four IF-300 casks in operation; one of these is owned by a utility company.

3. Nuclear Fuel Services, Inc. (NFS) has a legal-weight truck cask system (designated as NFS-4). The NFS-4 cask is capable of transporting one PWR fuel assembly or two BWR fuel assemblies. Two NFS-4 casks exist. However, the Nuclear Regulatory Commission suspended the Certificate of Compliance for operation of the NFS-4 casks in 1979.

4. Nuclear Assurance Corporation has a legal-weight truck cask system (designated NAC-1). The NAC-1 is identical in design to the NFS-4 cask and shares a common NRC Certificate of Compliance with NFS (the NFS-4). Three NAC-1 casks are owned by NAC and two NAC-1 casks are owned by a public utility. However, only three of these casks (two owned by NAC and one owned by a utility) are currently operable due to the show-cause order and these casks can only be operated dry on a derated basis of 2.5 kW per assembly decay heat output. In addition, NAC has designed and is seeking certification of a rail-sized, all-steel cask, designated the NAC-3. This cask would be capable of carrying 12 PWR assemblies or 32 BWR assemblies.

5. Transnuclear, Inc. (TNI) has two overweight truck cask systems (designated TN-8 and TN-9). The TN-8 is capable of transporting three PWR fuel assemblies and the TN-9 is capable of transporting seven BWR fuel assemblies. One TN-9 has been purchased by a utility company; this is the only one currently available in the United

States. TNI has under construction two each of the TN-8 and TN-9 casks for use in the United States by the end of 1980.

Currently there are 53 rail casks and 8 truck casks operating overseas as compared to 6 rail casks and 9 truck casks operating in the U.S.

The capacity of the casks in the United States for transporting spent fuel from reactor storage pools to an AFR storage facility has been estimated; results are set forth in Table VI-2. Capacity of transportation casks is conventionally cited in terms of metric tons of heavy metal (MTHM), which for LWR fuel is essentially equivalent to metric tons of uranium: Casks owned by utility companies have been included in the table because, for the purposes of this estimate, it was assumed that such casks would be used for shipment of spent fuel to AFR's by the cask owners.

From Table VI-2, it can be seen that the spent-fuel shipping casks currently available from commercial organizations engaged in providing such service have a capacity for transporting 511 MTHM/yr over average one-way shipping distances of 1,000 miles (with the rail casks having a capability of transporting 309 MTHM/yr of this total).

Utilizing existing casks, the transportation requirements imposed by the proposed AFR can be met until 1987. Beyond 1987, additional casks will be required to keep pace with AFR requirements. These requirements were described in Part V, and cumulative spent fuel shipments are shown in Table V-3.

With the steady operation of today's spent-fuel shipping cask inventory, additional casks could become available at a pace adequate to meet the later year demands.

A much higher capacity for transporting spent fuel will be required, once repositories are available to receive spent fuel, because shipments can be expected to be made from both reactor storage pools and AFR storage facilities. There are enough commercial suppliers of spent-fuel shipping casks and services to meet the demand for additional shipping casks.

Table VI-2. Capacity of Existing Licensed Spent-Fuel Shipping Casks

Cask	Average Transport Time (Days)[a]	Cask-Handling Crew Maintenance Time (Hrs)[b]	Average Time at Reactor (Days)[c]	Average Time at AFR (Days)[d]	Average Round-Trip Time (Days)	Number of Shipments per Year[e]	Average Capacity (MTU)[f]	Number of Casks Available[g]	Total Yearly Shipping Capacity (MTU)
NLI-1/2	2-1/2	13	2-1/2	1	6	50	.42	5	105
NAC-1	2-1/2	13	2-1/2	1	6	50	.42	3	63
NFS-4	2-1/2	13	2-1/2	1	6	50	.42	0	0
TN-9	6-1/2[h]	16	3	1	10-1/2	28	1.23	1	34
NLI-10/24	14	28	5	1-1/2	20-1/2	15	4.40	2	132
IF-300	14	36	6	2	22	14	3.16	4	177
								Total	511

[a] Based on 1,000-mile one-way shipments, 35 mph average highway speed, 6 mph average rail speed.

[b] Source: (Reference 4) G.H. Winsor, D.H. Faletti, and J.G. DeSteese, Opportunities to Increase the Productivity of Spent Fuel Shipping Casks, PNL-3017/UC-71 (Draft) Pacific Northwest Laboratory, Richland, WA, August 1979

[c] Source: (Reference 4) Adopted from G.H. Winsor, D.H. Faletti, and J.G. DeSteese, Opportunities to Increase the Productivity of Spent Fuel Shipping Casks, PNL-3017/UC-71 (Draft) Pacific Northwest Laboratory, Richland, WA, August 1979

[d] Assuming 24-hr, 7 days a week operation and 12-hr notification of the railroads.

[e] Assuming casks are available 300 days/year (an 82% availability factor).

[f] Assuming 1/3 BWR assemblies (0.176 MTU/assembly) and 2/3 PWR assemblies (.45 MTU/assembly).

[g] Includes all licensed casks, including utility-owned. Two NFS-4 and NAC-1 casks are not included due to NRC show cause order.

[h] Overweight permit restricts movement to daylight, weekday hours only.

The amounts of spent fuel to be transported and the number of casks required for the period 1997-2010 are shown in Table VI-3. These estimates are based on the following assumptions:

1. The first repository will be able to receive spent fuel in 1997. It will receive 1,800 MTU/year for the first 5 years of its operation and 6,000 MTU/year thereafter, until it reaches its capacity of 41,000 MTU.

2. Subsequent repositories will be brought on line at three-year intervals and will receive fuel on the same schedule as above, until their capacities of 69,000 MTU each are reached.

3. Transport of spent fuel in the time period 1997-2010 will be accomplished by rail and truck, with the distribution being 90% by rail and 10% by truck.

From Table VI-3, it can be seen that about 44 rail casks and 14 truck casks will be required by 1997. These will increase to a peak of 203 rail casks in 2005 and 43 truck casks in 2010.

Casks currently can be supplied by spent-fuel shipping organizations with the lead times shown in Table VI-4.

There are parts available at the present time for some cask systems that would reduce the time requirements necessary to build the first few additional casks needed. However, as a general rule, it must be assumed that it will require about 24 months lead time to effect delivery of a spent-fuel shipping cask after an order has been placed. Provided that casks are ordered on a timely basis, there does not appear to be any reason why all of the necessary casks cannot be provided from commercial sources to meet the needs set forth in Table VI-3.

Table VI-3. Spent Fuel Cask Requirements for Transport From Reactor and AFR Storage to Repositories[a] (1997-2010)

Year	Spent Fuel Transfers (1,000 MTU) Reactor to Repository	Reactor to AFR	AFR to Repository	Number of Truck Casks Needed	Number of Rail Casks Needed
1997	.9	1.7	-	14	44
1998	1.8	1.2	-	18	59
1999	1.8	1.8	-	20	67
2000	2.7	1.2	-	24	80
2001	3.6	0.6	-	27	93
2002	4.4	-	1.3	30	121
2003	4.6	-	4.1	31	162
2004	4.7	-	4.9	31	174
2005	4.9	-	6.8	33	203
2006	4.9	-	2.7	33	151
2007	6.6	-	-	34	156
2008	5.8	-	-	39	137
2009	6.2	-	-	42	147
2010	6.3	-	-	43	149

[a] Assumes the following:

(1) 2/3 PWR fuel, 1/3 BWR fuel (by weight).

(2) From reactors, 90% of fuel is shipped by rail, 10% is shipped by truck (by weight).

(3) NLI casks are used for reactor shipments and have the following annual payloads:

2,000 miles (one-way)	1,000 miles (one-way)
NLI 1/2 -- 15 MTU	NLI 1/2-----21 MTU
NLI 10/24 -- 38 MTU	NLI 10/24---66 MTU

(4) First repository starts 1997; capacity = 41,000 MTU
Second repository starts 2000; capacity = 69,000 MTU
Third repository starts 2003; capacity = 69,000 MTU

(5) Shipping distance -- 2,000 miles (one way) reactor to repository: 1,000 miles (one way) reactor to AFR storage, AFR storage to repository.

(6) From an AFR, 100% of fuel is moved by train utilizing NLI 10/24 shipping casks with a per cask capacity of 78 MTU/yr.

Table VI-4. Current Lead Times for Procurement of Spent-Fuel Shipping Casks from Commercial Services

Cask	Approximate Time Required to Make Cask Available from Date of Firm Contract for Its Use (Months)
NLI 1/2	24
NAC-1	12
NFS-4	12
NLI 10/24	24
IF-300	18
TN-8, 9	24

VI.F COSTS FOR AWAY-FROM-REACTOR STORAGE AND GEOLOGIC DISPOSAL OF SPENT FUEL

The costs to consumers of AFR storage and geologic disposal of spent fuel will be a small portion of the costs of electrical energy. In July 1978, the Department published preliminary estimates of the charge for spent fuel storage and disposal which were based on the early development and use of salt repositories for spent fuel disposal (5). This estimate of the cost of spent fuel storage and disposal amounted to about 0.5 mill/kWh for disposal only and to about 0.9 mill/kWh for both interim storage and disposal.

Since the issuance of this preliminary estimate, the Interagency Review Group (IRG) recommended to the President features of an Administration policy with respect to the long-term management of nuclear wastes and supporting programs to implement this policy. As a result of the recommendations of the IRG (6), the President has determined that, for interim planning purposes, the Department should locate and characterize a number of repository sites in a variety of different geologic environments with diverse rock types and should select one or more such sites for further development of full-scale repositories from four to five sites which have been carefully evaluated and found to be potentially suitable for repository location (7). This procedure will require additional research and development costs over a longer time period and a delay in repository deployment compared to the assumptions made in the July 1978 cost study.

The repository cost estimates and schedules discussed in Part III are being incorporated into an update of the spent fuel charge which would be imposed upon electric utilities. Preliminary calculations show that the storage and disposal charge and disposal-only charge will be somewhat higher than the 1978 Department of Energy estimate. However, the impact on the overall cost of electricity to the consumer is still expected to the small. This conclusion is supported by a recently-completed major international study, the International Nuclear Fuel Cycle Evaluation (INFCE). This study, involving representatives of forty nations, examined waste management and other aspects of seven alternative nuclear fuel cycles. Its conclusions noted that waste management costs could range between 0.8 and 2.0 mills/kWh among the various fuel cycles and for various geologic repository media (8). This compares favorably with 30-40 mills/kWh for delivered electricity.

References for Part VI

(1) Kaiser Engineers, Inc., Conceptual Design Report - National Waste Terminal Storage Repository for Storage of Spent Unreprocessed Fuel in a Bedded Salt Formation, Kaiser Engineers, Inc., Oakland, CA, December 1978

(2) C.R. Davis, correspondence dated 8 February 1978 to W.E. MacMath (Kaiser Engineers)

(3) C.D. Zerby, memorandum dated 5 January 1978 to Department of Energy, Oak Ridge Operations, Attn: J.J. Schreiber, "Maximum Throughput Rate of Spent Fuel Repository Based on Current Conceptual Design"

(4) G.H. Winsor, D.H. Faletti, and J.G. DeSteese, Opportunities to Increase the Productivity of Spent Fuel Shipping Casks, PNL-3017/UC-71 (Draft), Pacific Northwest Laboratory, Richland, WA, August 1979

(5) U.S. Department of Energy, Preliminary Estimates of the Charge for Spent Fuel Storage and Disposal Services, DOE/ET-0055, July 1978

(6) Interagency Review Group, Report to the President by the Interagency Review Group on Nuclear Waste Management, TID-29442, March 1979

(7) Presidential Message to Congress, 12 February 1980, "Comprehensive Radioactive Waste Management Program," Weekly Compilation of Presidential Documents, Vol. 16, No. 7

(8) International Nuclear Fuel Cycle Evaluation, INFCE Working Group 7, Waste Management and Disposal Summary, INFCE/WG.7/2 Rev. 2, p. 10, November 1979

VII. CONCLUSIONS

Based upon the foregoing Statement of Position, the Nuclear Regulatory Commission must find that it has confidence that:

1. Spent nuclear fuel from licensed facilities can be disposed of in a safe and environmentally acceptable manner;

2. The Federal Government's plans for establishing geologic repositories are an effective and reasonable means for developing a safe and environmentally acceptable disposal system;

3. Spent nuclear fuel from licensed facilities can be stored in a safe and environmentally acceptable manner on-site or off-site until disposal facilities are available;

4. Sufficient additional storage capacity for spent nuclear fuel from licensed facilities will be established; and

5. The disposal and interim storage systems for a spent nuclear fuel from licensed facilities will be integrated into an acceptable operating system.

Having made these findings, the Commission should promulgate a rule providing that the safety and environmental implications of spent nuclear fuel remaining on site after the anticipated expiration of the facility licenses involved need not be considered in individual facility licensing proceedings.

APPENDIX A

FOR IMMEDIATE RELEASE

Office of the White House Press Secretary

THE WHITE HOUSE

TO THE CONGRESS OF THE UNITED STATES:

Today I am establishing this Nation's first comprehensive radioactive waste management program. My paramount objective in managing nuclear wastes is to protect the health and safety of all Americans, both now and in the future. I share this responsibility with elected officials at all levels of our government. Our citizens have a deep concern that the beneficial uses of nuclear technology, including the generation of electricity, not be allowed to imperil public health or safety now or in the future.

For more than 30 years, radioactive wastes have been generated by programs for national defense, by the commercial nuclear power program, and by a variety of medical, industrial and research activities. Yet past governmental efforts to manage radioactive wastes have not been technically adequate. Moreover, they have failed to involve successfully the States, local governments, and the public in policy or program decisions. My actions today lay the foundation for both a technically superior program and a full cooperative Federal-State partnership to ensure public confidence in a waste management program.

My program is consistent with the broad consensus that has evolved from the efforts of the Interagency Review Group on Radioactive Waste Management (IRG) which I established. The IRG findings and analysis were comprehensive, thorough and widely reviewed by public, industry and citizen groups, State and local governments, and members of the Congress. Evaluations of the scientific and technical analyses were obtained through a broad and rigorous peer review by the scientific community. The final recommendations benefited from and reflect this input.

My objective is to establish a comprehensive program for the management of <u>all</u> types of radioactive wastes. My policies and programs establish mechanisms to ensure that elected officials and the public fully participate in waste decisions, and direct Federal departments and agencies to implement a waste management strategy which is safe, technically sound, conservative, and open to continuous public review. This approach will help ensure that we will reach our objective -- the safe storage and disposal of all forms of nuclear waste.

Our primary objective is to isolate existing and future radioactive waste from military and civilian activities from the biosphere and pose no significant threat to public health and safety. The responsibility for resolving military and civilian waste management problems shall not be deferred to future generations. The technical program must meet all relevant radiological protection criteria as well as all other applicable regulatory requirements. This effort must proceed regardless of future developments within the nuclear industry -- its future size, and resolution of specific fuel cycle and reactor design issues. The specific steps outlined below are each aimed at accomplishing this overall objective.

more

First, my Administration is committed to providing an effective role for State and local governments in the development and implementation of our nuclear waste management program. I am therefore taking the following actions:

- o By Executive Order, I am establishing a State Planning Council which will strengthen our intergovernmental relationships and help fulfill our joint responsibility to protect public health and safety in radioactive waste matters. I have asked Governor Riley of South Carolina to serve as Chairman of the Council. The Council will have a total of 19 members: 15 who are Governors or other elected officials, and 4 from the Executive departments and agencies. It will advise the Executive Branch and work with the Congress to address radioactive waste management issues, such as planning and siting, construction, and operation of facilities. I will submit legislation during this session to make the Council permanent.

- o In the past, States have not played an adequate part in the waste management planning process -- for example, in the evaluation and location of potential waste disposal sites. The States need better access to information and expanded opportunity to guide waste management planning. Our relationship with the States will be based on the principle of consultation and concurrence in the siting of high level waste repositories. Under the framework of consultation and concurrence, a host State will have a continuing role in Federal decisionmaking on the siting, design and construction of a high level waste repository. State consultation and concurrence, however, will lead to an acceptable solution to our waste disposal problem only if all the States participate as partners in the program I am putting forth. The safe disposal of radioactive waste, defense and commercial, is a national, not just a Federal, responsibility.

- o I am directing the Secretary of Energy to provide financial and technical assistance to States and other jurisdictions to facilitate the full participation of State and local government in review and licensing proceedings.

Second, for disposal of high level radioactive waste, I am adopting an interim planning strategy focused on the use of mined geologic repositories capable of accepting both waste from reprocessing and unreprocessed commercial spent fuel. An interim strategy is needed since final decisions on many steps which need to be taken should be preceded by a full environmental review under the National Environmental Policy Act. In its search for suitable sites for high level waste repositories, the Department of Energy has mounted an expanded and diversified program of geologic investigations that recognizes the importance of the interaction among geologic setting, repository host rock, waste form and other engineered barriers on a site-specific basis. Immediate attention will focus on research and development, and on locating and characterizing a number of potential repository sites in a variety of different geologic environments with diverse rock types. When four to five sites have been evaluated and found potentially suitable, one or more will be selected for further development as a licensed full-scale repository.

It is important to stress the following two points: First, because the suitability of a geologic disposal site can be verified only through detailed and time-consuming site specific evaluations, actual sites and their geologic environments must be carefully examined. Second, the development of a repository will proceed in a careful step-by-step manner. Experience and information gained at each phase will be reviewed and evaluated to determine if there is sufficient knowledge to proceed with the next stage of development. We should be ready to select the site for the first full-scale repository by about 1985 and have it operational by the mid-1990's. For reasons of economy, the first and subsequent repositories should accept both defense and commercial wastes.

Consistent with my decision to expand and diversify the Department of Energy's program of geologic investigation before selecting a specific site for repository development, I have decided that the Waste Isolation Pilot Plant project should be cancelled. This project is currently authorized for the unlicensed disposal of transuranic waste from our National defense program, and for research and development using high level defense waste. This project is inconsistent with my policy that all repositories for highly radioactive waste be licensed, and that they accept both defense and commercial wastes.

The site near Carlsbad, New Mexico, which was being considered for this project, will continue to be evaluated along with other sites in other parts of the country. If qualified, it will be reserved as one of several candidate sites for possible use as a licensed repository for defense and commercial high level wastes. My fiscal year 1981 budget contains funds in the commercial nuclear waste program for protection and continued investigation of the Carlsbad site. Finally, it is important that we take the time to compare the New Mexico site with other sites now under evaluation for the first waste repository.

Over the next five years, the Department of Energy will carry out an aggressive program of scientific and technical investigations to support waste solidification, packaging and repository design and construction including several experimental, retrievable emplacements in test facilities. This supporting research and development program will call upon the knowledge and experience of the Nation's very best people in science, engineering and other fields of learning and will include participation of universities, industry, and the government departments, agencies, and national laboratories.

Third, during the interim period before a disposal facility is available, waste must and will continue to be cared for safely. Management of defense waste is a Federal responsibility; the Department of Energy will ensure close and meticulous control over defense waste facilities which are vital to our national security. I am committed to maintaining safe interim storage of these wastes as long as necessary and to making adequate funding available for that purpose. We will also proceed with research and development at the various defense sites that will lead the processing, packaging, and ultimate transfer to a permanent repository of the high level and transuranic wastes from defense programs.

In contrast, storage of commercial spent fuel is primarily a responsibility of the utilities. I want to stress that interim spent fuel storage capacity is not an alternative to permanent disposal. However, adequate storage is necessary

more

(OVER)

until repositories are available. I urge the utility industry to continue to take all actions necessary to store spent fuel in a manner that will protect the public and ensure efficient and safe operation of power reactors. However, a limited amount of government storage capacity would provide flexibility to our national waste disposal program and an alternative for those utilities which are unable to expand their storage capabilities.

I reiterate the need for early enactment of my proposed spent nuclear fuel legislation. This proposal would authorize the Department of Energy to: (1) design, acquire or construct, and operate one or more away-from-reactor storage facilities, and (2) accept for storage, until permanent disposal facilities are available, domestic spent fuel, and a limited amount of foreign spent fuel in cases when such action would further our non-proliferation policy objectives. All costs of storage, including the cost of locating, constructing and operating permanent geologic repositories, will be recovered through fees paid by utilities and other users of the services and will ultimately be borne by those who benefit from the activities generating the wastes.

Fourth, I have directed the Department of Energy to work jointly with states, other government agencies, industry and other organizations, and the public, in developing national plans to establish regional disposal sites for commercial low level waste. We must work together to resolve the serious near-term problem of low level waste disposal. While this task is not inherently difficult from the standpoint of safety, it requires better planning and coordination. I endorse the actions being taken by the Nation's governors to tackle this problem and direct the Secretary of Energy to work with them in support of their effort.

Fifth, the Federal programs for regulating radioactive waste storage, transportation and disposal are a crucial component of our efforts to ensure the health and safety of Americans. Although the existing authorities and structures are basically sound, improvements must be made in several areas. The current authority of the Nuclear Regulatory Commission to license the disposal of high level waste and low level waste in commercial facilities should be extended to include spent fuel storage, and disposal of transuranic waste and non-defense low level waste in any new government facilities. I am directing the Environmental Protection Agency to consult with the Nuclear Regulatory Commission to resolve issues of overlapping jurisdiction and phasing of regulatory actions. They should also seek ways to speed up the promulgation of their safety regulations. I am also directing the Department of Transportation and the Environmental Protection Agency to improve both the efficiency of their regulatory activities and their relationships with other Federal agencies and state and local governments.

Sixth, it is essential that all aspects of the waste management program be conducted with the fullest possible disclosure to and participation by the public and the technical community. I am directing the departments and agencies to develop and improve mechanisms to ensure such participation and public involvement consistent with the need to protect national security information. The waste management program will be carried out in full compliance with the National Environmental Policy Act.

more

Seventh, because nuclear waste management is a problem shared by many other countries and decisions on waste management alternatives have nuclear proliferation implications, I will continue to encourage and support bilateral and multilateral efforts which advance both our technical capabilities and our understanding of spent fuel and waste management options, which are consistent with our non-proliferation policy.

In its role as lead agency for the management and disposal of radioactive wastes and with cooperation of the other relevant Federal agencies, the Department of Energy is preparing a detailed National Plan for Nuclear Waste Management to implement these policy guidelines and the other recommendations of the IRG. This Plan will provide a clear road map for all parties and will give the public an opportunity to review the entirety of our program. It will include specific program goals and milestones for all aspects of nuclear waste management. A draft of the comprehensive National Plan will be distributed by the Secretary of Energy later this year for public and Congressional review. The State Planning Council will be directly involved in the development of this plan.

The Nuclear Regulatory Commission now has underway an important proceeding to provide the Nation with its judgment on whether or not it has confidence that radioactive wastes produced by nuclear power reactors can and will be disposed of safely. I urge that the Nuclear Regulatory Commission do so in a thorough and timely manner and that it provide a full opportunity for public, technical and government agency participation.

Over the past two years as I have reviewed various aspects of the radioactive waste problem, the complexities and difficulties of the issues have become evident -- both from a technical and, more importantly, from an institutional and political perspective. However, based on the technical conclusions reached by the IRG, I am persuaded that the capability now exists to characterize and evaluate a number of geologic environments for use as repositories built with conventional mining technology. We have already made substantial progress and changes in our programs. With this comprehensive policy and its implementation through the FY 1981 budget and other actions, we will complete the task of reorienting our efforts in the right direction. Many citizens know and all must understand that this problem will be with us for many years. We must proceed steadily and with determination to resolve the remaining technical issues while ensuring full public participation and maintaining the full cooperation of all levels of government. We will act surely and without delay, but we will not compromise our technical or scientific standards out of haste. I look forward to working with the Congress and the states to implement this policy and build public confidence in the ability of the government to do what is required in this area to protect the health and safety of our citizens.

JIMMY CARTER

THE WHITE HOUSE

# # # # #

APPENDIX B
CURRENT EXPLORATION PROGRAMS

This appendix presents a detailed description of the current repository siting investigations of the NWTS Program, and it supplements the project summaries in Section II.D.5.

All figures, tables, references, and much of the text contained in Section II.D.5 are repeated in this appendix. This appendix is included to provide the interested reader with additional information and references for each of the exploration programs summarized in the body of the Statement. Descriptions are provided of status of investigations in:

B.1 - the Gulf Interior Region (salt domes)

B.2 - the Paradox Basin (bedded salt)

B.3 - the Permian Basin (bedded salt)

B.4 - the Salina Basin (bedded salt)

B.5 - the Department's Hanford Site (basalt)

B.6 - the Department's Nevada Test Site (volcanic tuff, argillite, granite, and alluvium), and

B.7 - the expanded NWTS national screening program (granite, shale, and geohydrologic environment)

The pertinent findings of the first six programs (B.1 - B.6) are discussed with respect to each of the four major factors of the natural systems as defined in Section II.D.2: i.e. geologic, hydrologic, tectonic, and resource factors.

Many geologic time periods are referred to in the Appendix by name. Figure B-1 presents a geologic time scale to enable the reader to associate the geologic periods with age in years (1).

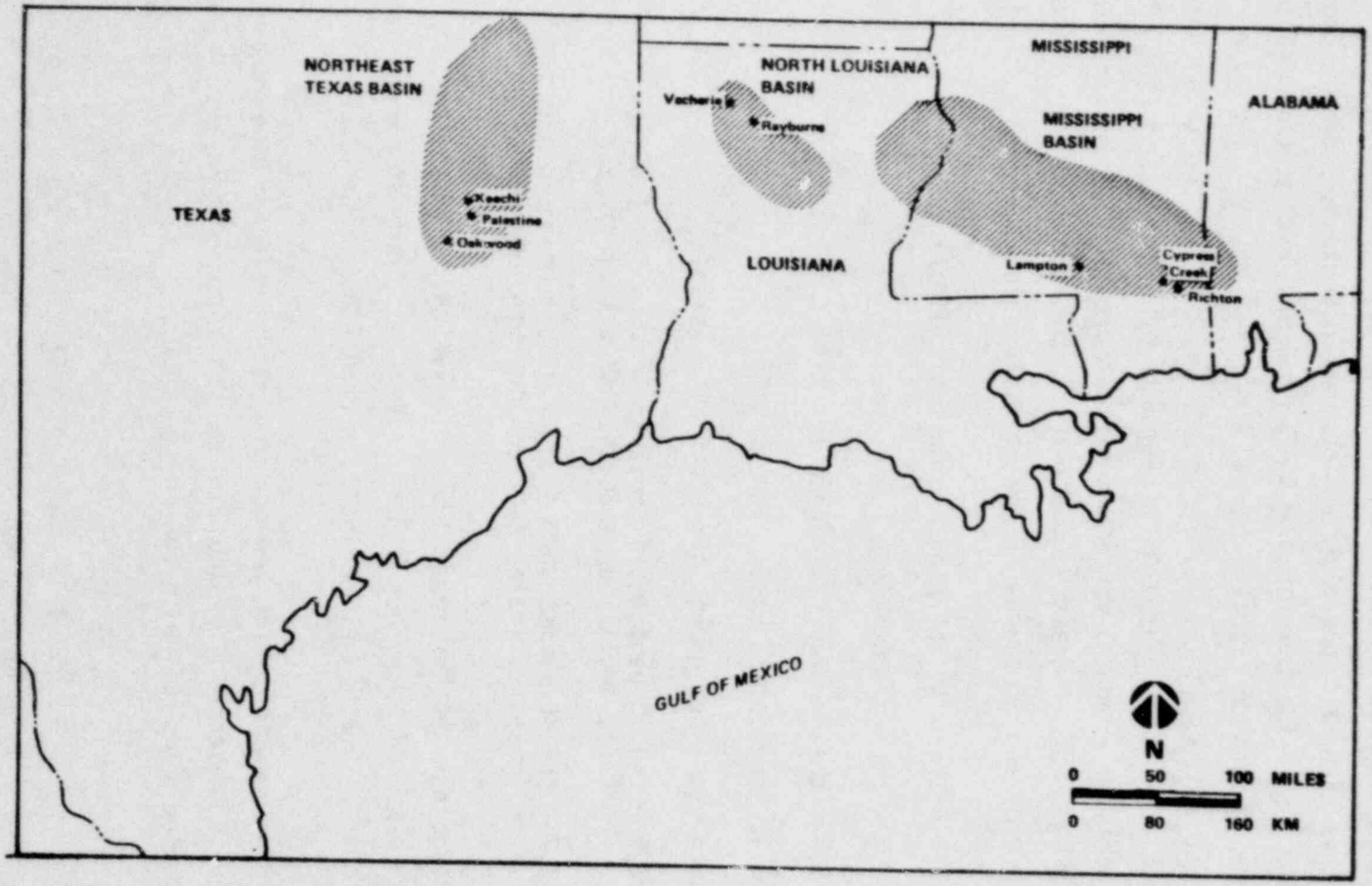


Figure B-1. Gulf Interior Region
Gulf Coast Salt Domes Recommended for Futher Study by the U.S. Geological Survey

B.1 Gulf Interior Region Salt Domes

B.1.1 Summary

The Gulf Coastal Plain of Texas, Louisiana, and Mississippi and adjacent offshore areas contain more than 500 salt domes, 263 of which are known or suspected to be on land. These interior domes were evaluated in 1963 by the U.S. Geological Survey, and 36 were identified as potentially acceptable for repository siting (2). The USGS screening criteria were (i) depth to salt of less than 2,000 ft and (ii) lack of previous use (oil, gas, sulfur, or brine mining). After the USGS study, the Department, with the participation of USGS district offices, selected 125 interior domes for detailed studies, which have provided important background data for current investigations (3-7).

Since 1978, interior domes have been evaluated by the Department in terms of geologic and other screening specifications (8-10). The three criteria that dominated the screening were (i) the top of the domes should be at depths of less than 915 m, (ii) domes should have cross-sectional areas of more than 1,000 acres, and (iii) domes should not have been used for hydrocarbon production or storage, or any other mineral-related activities. This latest study resulted in the selection of eight salt domes for the area study phase. These domes are distributed among the States of Louisiana, Mississippi, and Texas: Vacherie and Rayburns (Louisiana); Richton, Cypress Creek, and Lampton (Mississippi); and Oakwood, Keechi, and Palestine (Texas). The Palestine dome was dropped from further consideration in 1979 because of hydrologic uncertainties related to earlier solution mining (11). Fifteen karst-like collapse structures over the Palestine dome have been attributed to extensive brine production and solution collapse. Although the solution mining occurred from 1904 to 1937, three collapses have occurred since, one as recently as 1978.

Investigations under way at the remaining seven domes include hydrologic studies of the three sedimentary basins in which the domes occur (Figure B-2) as well as dome-specific geologic and hydrologic studies. Aquifers are being investigated to depths of 2 km in each basin by borehole pump tests. Geologic studies include regional and dome-specific field mapping

ERA	PERIOD	EPOCH	YEARS	
			DURATION	BEFORE THE PRESENT
CENOZOIC	Quaternary	Holocene	To Present	
		Pleistocene	1,000,000	1,000,000
	Tertiary	Pliocene	12,000,000	
		Miocene	12,000,000	
		Oligocene	11,000,000	
		Eocene	22,000,000	
		Paleocene	5,000,000	63,000,000
MESOZOIC	Cretaceous		72,000,000	135,000,000
	Jurassic		46,000,000	181,000,000
	Triassic		49,000,000	230,000,000
PALEOZOIC	Permian		50,000,000	280,000,000
	Pennsylvanian		30,000,000	310,000,000
	Mississippian		35,000,000	345,000,000
	Devonian		60,000,000	405,000,000
	Silurian		20,000,000	425,000,000
	Ordovician		75,000,000	500,000,000
	Cambrian		100,000,000	600,000,000
	PRECAMBRIAN			

Figure B-2. Geologic Time Scale

Source: (Reference 1) A. Holmes and L. Holmes, Principles of Physical Geology, 3rd ed., Halsted Press, New York, NY, 1978 with emphasis on evaluating Quaternary terraces, remote-sensing data, geophysical well logs, and deep seismic data. Understanding of dome locations is being refined by gravity surveys, high-resolution seismic reflection and refraction surveys, and borehole evaluations.

Drilling of a total of 34 deep exploratory holes in three States aggregating 89,189 ft, and drilling of additional intermediate-depth and shallow holes produced no evidence to disqualify any of the remaining seven domes, although several characteristics need careful evaluation against the siting criteria. Exploration is proceeding at a variable pace for all seven domes; only the Lampton dome in Mississippi and the Keechi dome in Texas have not been explored by drilling or by seismic methods.

Studies of hydrologic stability are centered on determining the resistance of salt masses to external dissolution. Current evidence suggests that salt domes have become, through geologic time, encapsulated in plastic clays or other impermeable minerals. These layers of impermeable sediments appear to have prevented the domes from dissolving since their formation 25 to 30 million years ago. Direct evidence for this hypothesis is found in oil company core samples and in geologic and drilling logs. The degree of continuity of the encapsulating clay needs to be determined. Preliminary calculations suggest long travel times for radionuclide migration to the accessible biosphere (at least 100,000 years). This is because thick sections of clay with a very low hydraulic conductivity (permeability) occur around and over the domes.

With regard to the tectonic stability of the domes, studies of Tertiary and older strata suggest that movement of the salt in the Gulf interior region ceased perhaps as long as 30 million years ago (4, 11). Assessments based on the thinning of sediments over the dome show a slow and declining rate of growth through geologic time, with values in the early Cenozoic well below 0.2 mm/yr (4). In addition, careful investigations of the stratification of Quaternary units in Louisiana have revealed no characteristics attributed to dome growth. All of the seven domes being investigated are considered to be tectonically stable. No capable faults are known to exist in the vicinity of the domes.

There is minor, and declining, oil production at the Oakwood and Cypress Creek domes. All domes have one resource exploration hole that penetrate to the salt. All domes have some degree of caprock.

In 1980 two or three domes will be recommended for further examination in the "location" study phase of the site-exploration process.

B.1.2 General Description

The Gulf interior region is a 200- to 600-mile-wide strip that extends inland from the Gulf of Mexico to the Mississippi Embayment (Figure B-2). Elevations within the region average several hundred feet above sea level, and drainage generally extends gulfward. The Mississippi River is within an alluvial valley, 25 to 125 miles wide, that extends southward to the Gulf of Mexico along the axis of the Mississippi Embayment. The area is characterized by broad valleys and a generally subdued terrain.

The Gulf interior basin was initially formed in the Late Triassic Period by block faulting and rifting of the continental crust, accompanied by basic igneous activity. Differential subsidence during the Early Jurassic resulted in isolated basins that were centers for the accumulation of thick salt deposits.

Subsequent marine sedimentation continued until the middle Cretaceous, when it was interrupted by a general emergence of the land. Resumption of marine sedimentation occurred in the Late Cretaceous, when the sea transgressed to southern Illinois, forming the Mississippi Embayment. Continental emergence started during the Cenozoic and extended progressively gulfward.

The geosyncline depositional patterns in the region consisted of alternative periods of submergence characterized by the deposition of limestones, clastics, or scattered evaporites and emergence characterized by deposition of shaley deltaic materials. The resulting deposits are sufficiently widespread to allow stratigraphic correlation throughout the area (Figure B-2).

Structural movement within the basins occurred in the Louann Salt (Figure B-3) due to the weight of overlying sediments and density contrasts between the salt and adjacent geologic units (12). Salt movement began

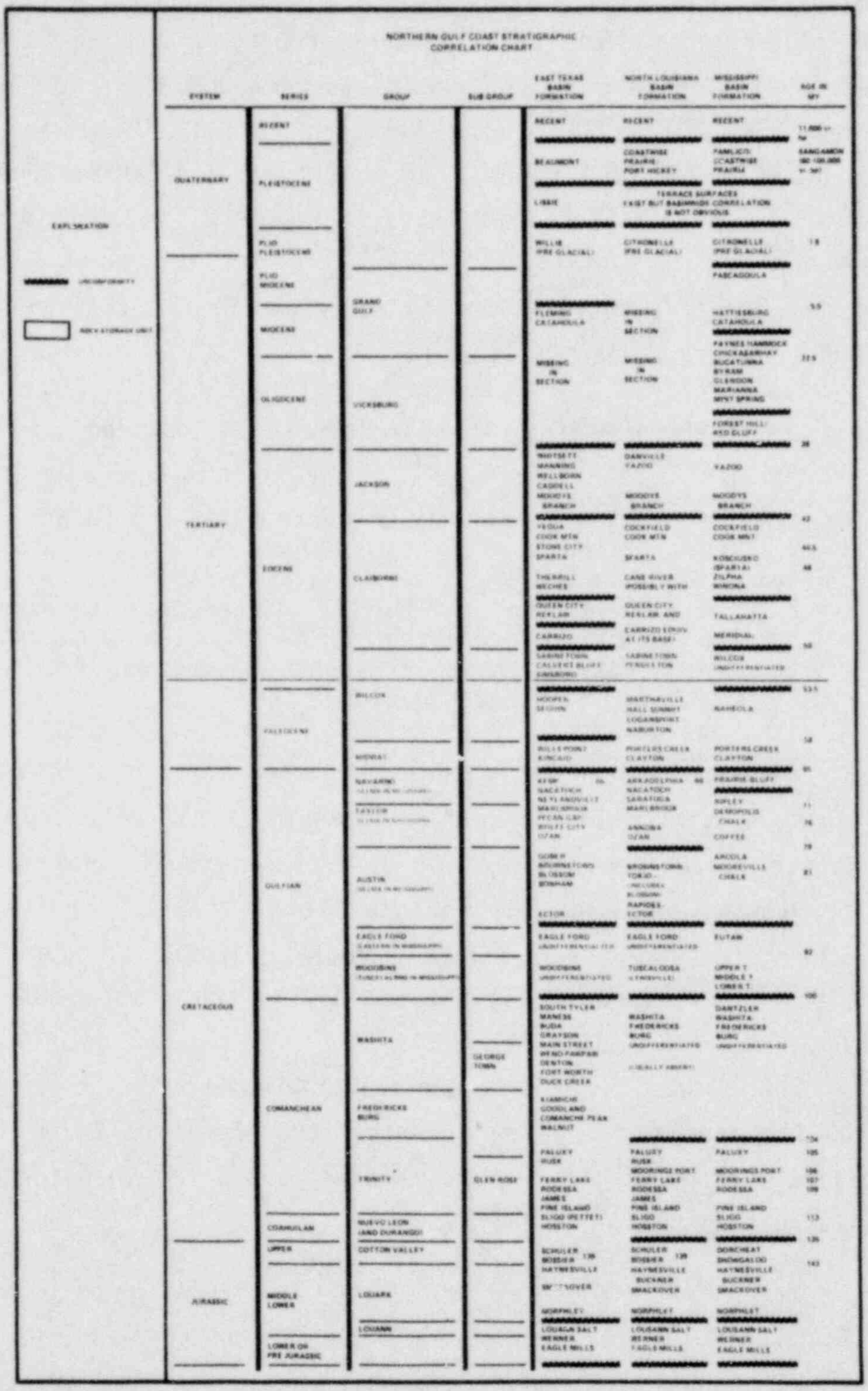


Figure B-3. Strat:graphic Correlation Chart for the Northern Gulf Coast.

in the Late Jurassic by the formation of ridges, which eventually developed into "diapiric* domes" as additional sediment loading occurred over the thick salt strata. Salt movement and diapirism in the interior salt basins climaxed during the Mesozoic Period. Diapiric growth was greatest in areas of maximum deposition. Salt diapirs in the Gulf interior region are in pillar-and-piercement structures, with related faulting and folding.

In general, most geologists agree on three points concerning the origin of Gulf Coast salt diapirs (12):

1. Salt in diapiric structures is derived from bedded sedimentary salt.

2. Salt in diapiric structures moved by plastic deformation.

3. The combination of density differences between the salt and overlying sediments, and sediment consolidation due to sediment loading was sufficient to cause dome development.

Good-quality ground water is present throughout the Gulf interior region and is used extensively for domestic, municipal, and industrial purposes. Important aquifers in the region include the Wilcox-Carrizo units, the Sparta Formation, the Queen City Sand, the Cockfield Formation, Miocene sands, and Pleistocene-to-Recent alluvial valley deposits (Figure B-3). In general, the base of fresh water is between 500 to 1000 feet below sea level (MSL) throughout the region (13, 14).

The Midway clay is a confining unit because of its thickness (500 to 1,500 ft) and its very low permeability. Aquifers below the Midway Group are saline throughout the region. Whether clays in other formations are confining units has not yet been determined.

---

* The term "diapir" applies to more than salt domes. The core of a diapir can be salt, clay, sand, serpentine, or a variety of other materials that can assume many shapes, like those of a tear drop or a mushroom.

The Gulf interior region is an area of low seismicity, classified as seismic risk zone 0 or 1 (15). The seismic implications for repository siting in the Gulf interior region can be summarized as follows:

1. The low level of historical earthquake activity, when considered alone, indicates either a low level of current tectonic activity or a tectonic activity associated principally with the aseismic release of elastic strain energy.

2. There is no clear correlation between geologic structures and regional earthquake activity. The Department is delineating seismo-tectonic provinces to facilitate the estimation of seismic risk.

3. Future vibratory ground-motion levels from local and distant earthquakes are expected to be low (16).

Specific descriptions of each of the domes in the Gulf interior region are presented in the sections that follow.

B.1.3 Keechi Dome, Anderson County, Texas

B.1.3.1 Geologic Factors--Keechi Dome

The domal structure at Keechi is expressed as a topographic low. Topographic relief over the area is on the order of 30 to 40 ft. Keechi Creek traverses the central dome area and flows to the south into the Trinity River.

The salt at the Keechi dome extends to within 435 ft of the surface. A caprock is believed to drape over the dome and vary in thickness from 16 to 758 ft. The horizontal cross-sectional area is 1,180 acres at 2,000 ft below MSL* and 2,069 acres at 3,000 ft below MSL, making it one of the smallest of the domes being considered.

Formations as old as the Cretaceous Taylor and Navarro Groups have been pushed to the surface near the center of the dome. Surface sediments generally become progressively younger away from the dome. These outcrop

---

*Mean sea level

patterns are modified by radial faulting associated with ancient dome growth. Detailed evaluations of faulting in the area need to be completed before the potential impact of these features on possible site suitability and repository design can be assessed.

B.1.3.2 Hydrologic Factors--Keechi Dome

At the Keechi dome, the Wilcox Group fresh water aquifer is partially shielded from the salt mass by the Midway shale, which crops out at the surface and drapes over portions of the salt. However, at this time, data are insufficient for a complete quantitative subsurface hydrologic characterization.

Much of the effort in subsurface hydrology has been devoted to obtaining a preliminary estimate of the time that would be required for ground water travel from within the dome to the accessible biosphere. Preliminary calculations suggest very long travel times (at least 100,000 years). The computations were based on observed field data when feasible; however, numerous simplifying assumptions were made. No estimate of radioisotope travel time including sorption potential along the flow paths has been made.

B.1.3.3 Tectonic Factors--Keechi Dome

Aside from the faults associated with ancient doming, the faults nearest to the Keechi dome are 9 to 14 miles away. The most recent significant fault activity is believed to have occurred more than a million years ago. Seismicity along the faults is very low. The Texas Bureau of Economic Geology believes the faults are not tectonic (16). The supradomal and radial faults at Keechi are related to the growth of the dome.

B.1.3.4 Resource Factors--Keechi Dome

No nonevaporate mineral deposits exist on or adjacent to the Keechi dome. The nearest oil fields are 5 miles away (17).

B.1.4 Oakwood Dome, Freestone and Leon Counties, Texas

B.1.4.1 Geologic Factors--Oakwood Dome

Topographically, the Oakwood dome is characterized by low rolling hills with relief on the order of 70 ft. The salt extends to within 1,016 ft of the surface. Borehole and gravity data indicate that Oakwood Dome is circular, with an overhang on all sides giving the appearance of a mushroom. The base of the overhang is 4,000 to 5,000 ft below the surface. A caprock of anhydrite and sand, encountered in numerous wells penetrating the overhang, is 20 to 837 ft thick; it extends over the dome and down to the overhang. Beneath the overhang, "gouge", abraded material occurring between the walls of a fault, has been identified. The horizontal cross-sectional area of the dome is 2,785 acres at 2,000 ft below MSL and 2,613 acres at 3,000 ft below MSL.

The surface formations over the dome have recently been identified as the Queen City Formation, although earlier reports have suggested that sediments as young as the Cook Mountain Formation are present.

Local units exhibit structural discord, radial faulting, plus thinning and thickening as a result of domal growth.

B.1.4.2 Hydrologic Factors--Oakwood Dome

At Oakwood, the Wilcox Group fresh water aquifer is adjacent to the dome at approximately 1,500 ft below the surface and is believed to extend across the top of the dome. The Midway shale exists at the probable repository elevation. The Eagle Ford and Woodbine Groups lie in contact with the salt stock just under the overhang at a depth of 5,000 to 7,000 ft below the surface. At this time, data are insufficient for a complete quantitative subsurface hydrologic characterization, although preliminary estimates indicate ground water travel times to the biosphere of at least 100,000 years.

B.1.4.3 Tectonic Factors--Oakwood Dome

The regional faults nearest to the Oakwood dome are 5 to 15 miles away. The most recent significant activity in these features occurred more than a million years ago (16). The seismicity of the faults is considered to be insignificant, and it is believed that the formation is not basement related (16). Supradomal and radial faults at Oakwood are related to the growth of the dome.

B.1.4.4 Resource Factors--Oakwood Dome

Fifty to sixty oil wells have been drilled in the vicinity of the Oakwood dome. Oil and gas production has been established under the overhang from wells drilled through the overhang into the Woodbine Formation. During the last 5 years, production has been declining, however.

In 1975, four wells drilled into the Woodbine produced only about 10,000 bbl of oil and 3.5 x 10^9 ft^3 of gas. Other wells around the dome are plugged and abandoned, but the effectiveness of the plugs has not been determined. No other production exists within 2.5 miles of the dome, but some production does exist in Red Oak Fiel[illegible] to 5 miles to the southwest. No other mineral resource deposits have been developed in the vicinity of the dome (17).

B.1.5 Vacherie Dome, Webster and Bienville Parishes, Louisiana

B.1.5.1 Geologic Factors--Vacherie Dome

The terrain at Vacherie is dominated by a central depression encircled by hills, with a relief of more than 200 ft. Bashaway Creek occupies the low central portion of the dome and drains to the east into Black Lake Bayou, a tributary of the Red River.

The 777-ft depth to the shallowest salt has been determined by drilling. The thickness of caprock ranges from 79 to 273 ft. The northwest-trending dome is 3.5 to 4 miles long and approximately 2 miles wide at the center. Structural contours based on gravity data show the cross-sectional area of salt to be 2,286 acres at 2,000 ft below MSL and 2,879 acres at 3,000 ft below MSL.

Vacherie lies in an outcrop area of the Sparta Formation. The Cane River and Wilcox Formations are concentric inliers, exposed in a somewhat elliptical fashion around the dome. The central depression of Vacherie is covered by flood-plain deposits and the terrace sediments of Bashaway Creek (4).

The subsurface geology of Vacherie is complicated by faulting, which is responsible for a series of horst and graben structures in Tertiary strata above the salt. The sediments surrounding the dome dip away from it. Radial faults are believed to exist at Vacherie, but current data are insufficient to support this hypothesis.

B.1.5.2 Hydrologic Factors--Vacherie Dome

At the Vacherie dome, the Wilcox aquifer appears to be separated from the salt by the confining Midway clay, but it may rest on the caprock in several of the down-thrown faults blocks on top of the dome. The data available at present are insufficient for a complete quantitative subsurface hydrologic characterization.

B.1.5.3 Tectonic Factors--Vacherie Dome

The regional faults nearest to the Vacherie dome are in the Hosston Zone, 50 miles away. The most recent significant activity in the Hosston faults occurred before mid-Tertiary time, some 35 million years ago. The U.S. Geological Survey has described an inferred northwest-southeast fault that passes near both Vacherie and Rayburn's domes (2), but its existence has not been confirmed by field data. The seismicity of the Hosston faults is considered to be insignificant. Supradomal and possibly radial faults at Vacherie are believed to be related to dome growth.

B.1.5.4 Resource Factors--Vacherie Dome

Several petroleum exploratory wells have been drilled on the flanks of the Vacherie dome. No oil or gas has been found, suggesting that Vacherie has little potential for hydrocarbon development. There are currently no active well sites on the dome; however, several producing wells occur in the Ada Field, and several more wells are being drilled 2 to 3 miles to the northeast.

Lignite development potential in the vicinity of the dome is fair to good. No present nor past brine production is known to have occurred on Vacherie. There is some minor development of sand and gravel deposits in the vicinity of the dome. No other mineral resources have been produced in the vicinity (17).

B.1.6 Rayburn's Dome, Bienville Parish, Louisiana

B.1.6.1 Geologic Factors--Rayburn's Dome

Rayburn's Dome is encircled by low hills with a central saline marsh that drains into Fouse Bayou, a tributary of the Dugdomona River. Topographic relief in the vicinity of Rayburn's Dome is about 60 ft.

The depth to salt is 115 ft. Caprock is 5 to 88 ft thick and occurs 12 to 19 ft below the surface. The dome is slightly elongated in the northwest-southeast direction. Gravity models indicate the presence of a salt overhang between 2,000 and 4,000 ft below MSL. The horizontal cross-sectional area of the salt, as determined by gravity and well data, is 1,307 and 1,685 acres at 2,000 and 3,000 ft below MSL, respectively.

Quaternary deposits cover the low central portion of the dome (4). Upper Cretaceous sediments are exposed in outcrops east of the central saline marsh. From limited exposures it appears that the Midway, the Cane River, and the Sparta Formations form concentric outcrop bands surrounding most of the dome. There is some evidence for radial and concentrate faulting related to dome growth. In the subsurface, the flanking Upper Cretaceous and Tertiary strata are steeply inclined and dip away from the dome.

B.1.6.2 Hydrologic Factors--Rayburn's Dome

At Rayburn's dome, sparse well data indicate that the salt has penetrated sediments as young as the Wilcox aquifers. The Midway Group aquitard is thought to sheath the salt, but the data are inconclusive. Field data are insufficient to characterize quantitatively the subsurface hydrology.

B.1.6.3 Tectonic Factors--Rayburn's Dome

The regional faults nearest to Rayburn's dome occur in the Hosston Zone, 72 miles away.

B.1.6.4 Resource Factors--Rayburn's Dome

Several oil and gas exploratory wells have been drilled on and around Rayburn's dome; however, all were dry holes. The nearest petroleum production is from the Danville Field, which is approximately 2 miles southwest. The Liberty Hill and Lucky gas fields are about 3.5 miles north and northwest of the dome's center.

An abandoned limestone quarry in the Saratoga Chalk is located over the dome. Small amounts of brine were produced from the dome in the Civil War era, and minor solution mining occurred in the 1940's. There is a potential for the development of sand and gravel deposits near the dome, but no producing pits or quarries have been identified. Lignite development potential is fair to good nearby. No other mineral resouces are known to have been produced in the vicinity (17).

B.1.7 Richton Dome, Perry County, Mississippi

B.1.7.1 Geologic Factors--Richton Dome

The Richton dome is located beneath the drainage divide between the valleys of Bogue Homo and Thompson Creek, tributaries of the Leaf River. Surface elevations range between 290 and 160 ft on the dissected divide over the dome.

The shallowest caprock and salt at the Richton dome were penetrated at depths of 497 and 722 ft, respectively. Borehole data show that the dome is relatively flat-topped, with an overhang on the northern end. The horizontal cross-sectional area of salt is 4,621 and 4,114 acres at 2,000 and

3,000 ft below MSL, respectively. Richton is one of the largest of the domes under consideration.

Geologic mapping by the University oı Southern Mississippi has shown that the Miocene Hattiesburg Formation and the Plio-Pleistocene Citronelle Formation crop out above the dome. Quaternary and Recent alluvial deposits associated with Bogue Homo and Thompson Creek cross the western and eastern flanks.

B.1.7.2 Hydrologic Factors--Richton Dome

At this time, data are insufficient for a quantitative subsurface hydrologic characterization.

B.1.7.3 Tectonic Factors--Richton Dome

The regional faults nearest to the Richton dome are faults in the Pickens-Gilbertown Fault Zone (12). The most recent significant activity of the Pickens-Gibbertown Fault Zone occurred in Miocene time, about 11 millio: years ago. The seismicity of the faults is considered to be insignificant. Faults related to dome growth have been discovered adjacent to the dome.

B.1.7.4 Resource Factors--Richton Dome

Petroleum investigation on the flanks of the Richton dome has not identified significant quantities of petroleum in the uplifted strata. On the north flank of the dome beneath the overhang, an oil "show" was reported in a 100-ft section of Lower Cretaceous sand. The sand, which is more than 12,000 ft deep, was shown to be noncommercial by extensive testing. The well was subsequently abandoned. The oil-production area nearest the dome is about 2.5 miles north-northwest and about 2.7 miles south-southeast.

The potential for caprock sulfur deposits at Richton was investigated, unsuccessfully, in the 1940's. Thirty-one unsuccessful sulfur test wells were drilled into the caprock, eight of which penetrated salt. Active and abandoned pits are present over the dome, and a few sand and gravel and clay pits are located in the flood plains of Bogue Homo and Thompson Creek. There is fair potential for production of hydrogen sulfide gas. No other mineral resources have been identified or produced in the vicinity (17).

B.1.8 Lampton Dome, Marion County, Mississippi

B.1.8.1 Geologic Factors--Lampton Dome

The Lampton dome lies beneath an east-west trending, flat-topped divide between Upper Little Creek and Lower Little Prong Creek, tributaries of the Pearl River. Elevations on the ridge vary from 370 to 180 ft.

The salt is 1,646 ft below the surface, and the caprock ranges in thickness from 37 to 262 ft. Borehole and gravity data show the dome to be conical, with no overhang. The gravity data suggest that the horizontal area of the salt is about 1,075 and 1,236 acres at 2,000 and 3,000 ft below MSL, respectively.

Surface geologic mapping shows that the Miocene Pascagoula and Hattiesburg Formations are the youngest geologic units exposed over much of the area above Lampton Dome. The Plio-Pleistocene Citronelle Formation covers the higher hills east of the dome and may cap the ridge beneath the dome. Interpretation of the borehole data indicates that the dome penetrates the Miocene Catahoula Formation. The basal sedimentary unit which is interpreted to cross the dome is the Tatum Limestone Member of the Miocene Catahoula Sandstone.

B.1.8.2 Hydrologic Factors--Lampton Dome

The flanks of the Lampton dome are known from only a few wells that indicate steeply dipping bedding and diapiric shale in contact with the dome. It is hypothesized (151) that this shale may be protecting the dome from dissolution because no brine has been found in overlying sands. At this time, data are insufficient for a quantitative subsurface hydrologic characterization.

B.1.8.3 Tectonic Factors--Lampton Dome

The nearest regional faults are about 75 miles away. The most recent activity in the faults occurred in Miocene time, about 11 million years ago. The seismicity of the faults is considered to be insignificant. No dome-growth related faults have been discovered.

B.1.8.4 Resource Factors--Lampton Dome

The nearest oil- or gas-producing area is 4.5 miles southwest at Hub Field. Several exploratory wells have been drilled to investigate the petroleum potential of the dome, but none has been productive. Exploration in the 1940's for caprock sulfur deposits was also unsuccessful. Eight unsuccessful sulfur and petroleum exploration wells were drilled over the dome. There is a fair potential for hydrogen sulfide gas. No other mineral resources, with the exception of sand and gravel, have been produced in the vicinity (17).

B.1.9 Cypress Creek Dome, Perry County, Mississippi

B.1.9.1 Geologic Factors--Cypress Creek Dome

The Cypress Creek dome lies beneath a broad swamp at the headwaters of Cypress Creek, a tributary of Black Creek and, eventually, the Pascagoula River. Surface elevations over the dome vary from 180 to 270 ft.

Gravity and well data show an overhang on the north, east, and southwest flanks of the dome. Shallowest salt occurs at a depth of 1,320 ft. Anhydrite caprock is approximately 200 ft thick. Well and gravity data show the horizontal cross-sectional area of salt to be 2,598 acres at 2,000 ft below MSL and 2,249 acres at 3,000 ft below MSL.

The Miocene Hattiesburg Formation lies immediately beneath alluvium and is also found at the surface north and east of the dome. The Plio-Pleistocene Citronelle Formation crops out over much of the surface immediately over the dome according to recent mapping by the University of Southern Mississippi.

B.1.9.2 Hydrologic Factors--Cypress Creek Dome

At this time, ata are insufficient for a quantitative subsurface hydrologic characteriza on.

B.1.9.3 Tectonic Factors--Cypress Creek Dome

The nearest regional faults are related to the Jackson-Mobile Graben, 56 miles away. The most recent significant activity along the faults occurred in Miocene time, about 11 million years ago. The seismicity of the faults is considered to be insignificant. No dome-growth related faults have been interpreted.

B.1.9.4 Resource Factors--Cypress Creek Dome

Nine petroleum exploration wells have been drilled in the vicinity of the dome, two in the mid-1930's and the remaining seven since 1972; the latter were drilled to test the structure on the flanks on the dome. Four of these wells produced oil and gas, and three still remain productive. Production has been from beneath the overhang on the north flank.

With the exception of these wells, the nearest production is about 9 miles northwest, at Glazier Field. Hydrogen sulfide gas potential is fair. No other mineral resources, except sand or gravel, have been produced in the vicinity (17).

B.2 Paradox Basin

B.2.1 Summary

The Paradox basin in southeastern Utah and southwestern Colorado is a part of the Colorado Plateaus, an area of high plateaus and deeply incised canyons. The extensive flat-lying sedimentary rocks are sometimes folded into structural upwarps and occasionally interrupted by Tertiary volcanoes, volcanic necks, and igneous intrusions. Approximately 12,000 square miles of the Paradox basin are underlain by the bedded salt of the Paradox

Formation, which was deposited some 300 million years ago during Pennsylvanian time. More than 25 salt layers, separated by interbeds of shale, carbonates, and anhydrite, are present in some parts of the basin. The Paradox basin is one of the salt regions identified in a national screening as having potential for the eventual siting of a waste repository (18). Evaluation of the basin was begun in 1972 by the U.S. Geological Survey (19).

Existing information on the Paradox basin is not yet sufficient to make decisions concerning the suitability of any parts of the region for a repository. The information is quite sufficient, however, for identifying and guiding subsequent investigations. Chief among these are to acquire much more and better data on the hydraulic and chemical characteristics of the ground water flow systems, Quaternary history, and present-day seismicity. These and other investigations are now under way in the basin, in general, and, in some detail, in the four areas of the basin discussed below.

From existing information, a regional characterization (19) and screening of the Paradox basin have been completed. On the basis of a number of geologic factors and specifications (e.g., thickness of salt, depth to salt, and location of mapped faults), the basin was divided into several areas. The geologic screening results were then combined with environmental screening maps to identify four areas recommended for further evaluation in the "area" level study phase (see Figure B-4). One of the four areas, Salt Valley, in Grand County, Utah, had been previously identified by the U.S. Geological Survey (19). The other three areas, Gibson Dome, Elk Ridge, and Lisbon Valley, are in San Juan County, Utah. The results of geologic field and literature studies now in progress will be used to compare the characteristics of the four areas as a part of the process of deciding whether locations in one or more of the four areas will be recommended for further investigation (20).

In the near future, at least two deep holes, one in the Gibson Dome area and one in the Elk Ridge area, will be cored, logged, and extensively tested. Samples of fluids recovered during testing operations and samples of cores will be analyzed at the site and in the laboratory for water content of salt, mineral composition, kerogen and hydrocarbon content, and hydrology. A seismic reflection line is planned at Elk Ridge. Surface electromagnetic surveys will be conducted at Gibson Dome and Elk Ridge. Hole-to-

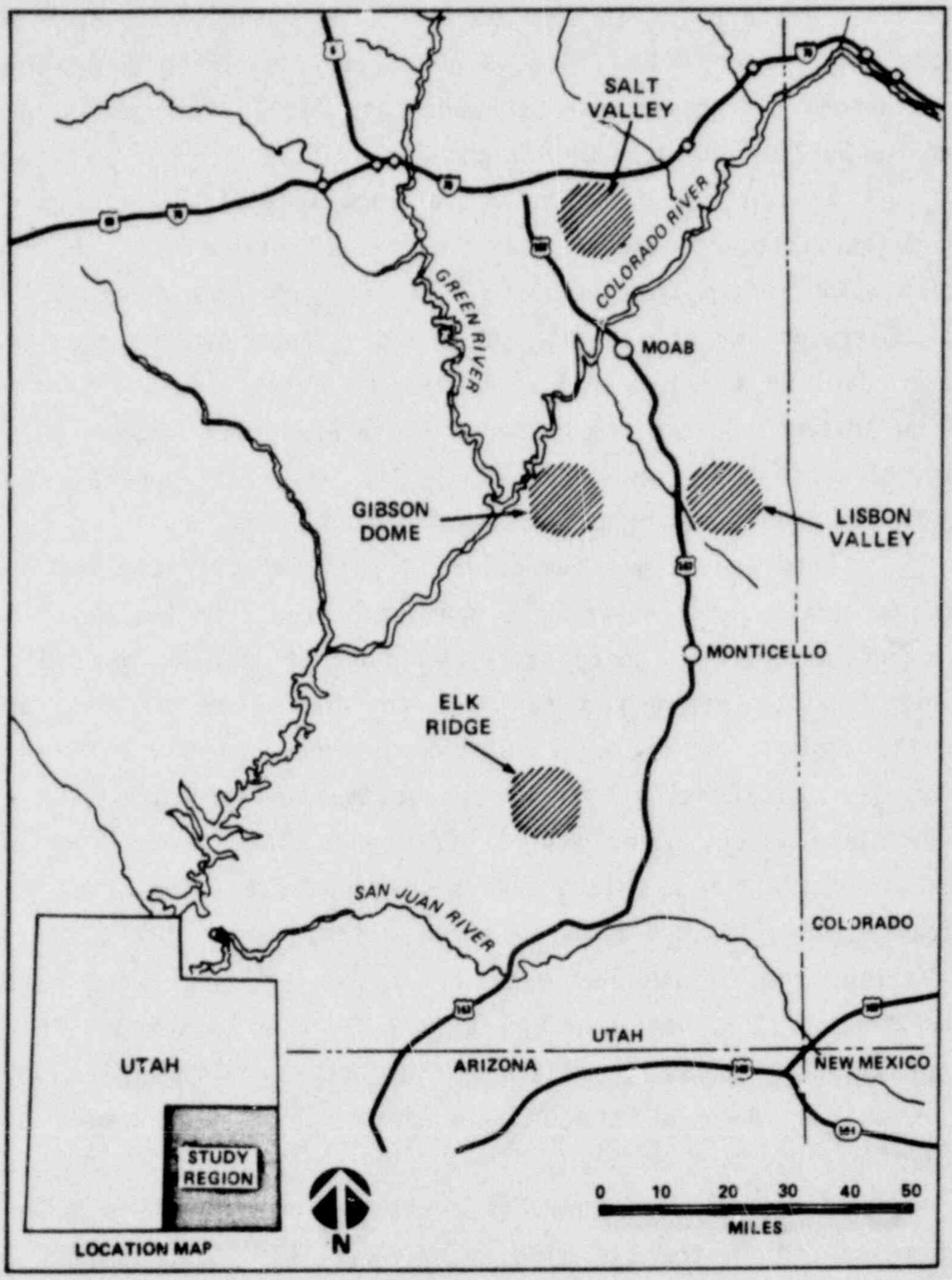


Figure B-4. Areas of the Paradox Basin Identified for Further Evaluation

Source: (Reference 19) R.J. Hite and S.W. Lohman, Geologic Appraisal of Paradox Basin Salt Deposits for Waste Emplacement, Open File Report 4339-6, U.S. Geological Survey, 1973 surface electrical resistivity surveys will be run in the vicinity of the planned deep drill holes. Data collected by a 24-station microseismic network over a period of 6 months are being analyzed.

Area-level field investigations for Lisbon Valley will be much less extensive than those in the other three areas because of the large amount of information amassed earlier for mining activities in this area. The potentially unfavorable characteristics of this area include the near proximity of a producing oil field and of extensive uranium deposits and exploration. Moreover, the area is known to be much more geologically complex than the Gibson Dome and Elk Ridge areas but probably less complex than the Salt Valley area.

Preliminary results from investigations conducted to date indicate that bedded salt layers of sufficient volume are present at suitable depths in the Paradox Basin. Historically, many earthquakes with Richter magnitudes exceeding 1.0 have occurred within 200 miles from the basin, but only 17 were in the basin itself. The largest had a Richter magnitude of 4.3 and was located near the basin margin. Potential resource-conflict and ground water flow system evaluations are in progress.

B.2.2 Geologic Factors--Paradox Basin

Geologic factors are being evaluated by examining existing surface and subsurface information, which is being supplemented by additional drilling and field geologic methods (21, 22).

The Paradox basin is a Paleozoic asymmetrical depositional trough. Underlying the Pennsylvanian evaporites of interest are marine deposits, generally carbonates and shales, of Cambrian and Mississippian age. No Silurian and Devonian rocks are present in the basin.

Up to 15,000 ft of Pennsylvanian and Permian sediments are present in the basin. The Permian sediments are primarily clastic. During the Pennsylvanian evaporitic stage of deposition, salt, carbonates, anhydrite, and black shale were deposited cyclically in the central part of the basin, but carbonate deposition predominated on the margins of the basin.

Mesozoic strata consist mostly of a thick marine shale and nonmarine clastics. The youngest rock exposed in the basin is a Cretaceous shale.

The regional structure of the basin is a homocline that dips at very low angles to the north and northeast. Superimposed on the regional structure are smaller structures--anticlines with very-low dipping flanks; steeply dipping monoclines; salt anticlines, both diapiric and nondiapiric; and Tertiary igneous intrusives that caused localized uplift and doming.

Three holes, two drilled to a total depth of about 1,250 ft and the third nearly continuously cored to a total depth of over 4,000 ft, have been completed at Salt Valley. Seismic refraction, electromagnetic, and hole-to-surface electrical resistivity surveys have been completed, as have two crosshole vertical seismic surveys to determine whether this "off-the-shelf" technique can be used to map the position and attitude of the anhydrite-carbonate-shale interbeds in the thick salt core of the Salt Valley anticline. Analysis of these data is not yet completed. Similar exploratory investigations are planned or under way in other areas of the basin.

B.2.3 Hydrologic Factors--Paradox Basin

In addition to an evaluation of existing ground water information, hydrologic factors are being investigated by both deep and shallow drilling and testing in strata below and above the salt beds of interest (20). Hydrologic testing in three pairs of shallow holes, each approximately 600 ft deep and bottomed at or near the top of salt, is under way at Salt Valley.

Analysis of field data and 4,000-ft of core from Salt Valley has not been completed. Preliminary results indicate that Salt Valley is potentially favorable hydrologically because of the very low rates and quantities of ground water flow in the caprock. Very long periods of time would thus be required for water to circulate through the caprock. This conclusion is supported by the age (more than 36,000 years) of a sample of water from the caprock, dated by the carbon-14 technique (22).

The subsurface hydrologic systems are influenced by structural and topographic elements in and bordering the Paradox basin. The Uncompahgre and Monument uplifts control the general pattern of ground water movements in and through the basin. The deeply incised terrain creates localized departures from the regional flow pattern (21).

Recharge to aquifers occurs mainly on the west flank of the San Juan Mountains and the west side of the Uncompahgre uplift. Recharge also occurs on and near the Monument uplift, the Abajo Mountains, the La Sal Mountains, and the Sage Plain (21).

Discharge from strata above the Paradox Formation occurs along the canyon walls of incised river channels. Wells into very shallow strata are also points of discharge in the region. The strata below the Paradox Formation do not crop out in the basin, and unless there is an upward movement of water into the overlying rocks, discharge from these strata would, in general, be to the south or southwest and outside the Paradox basin (21). Sufficient data have not yet been acquired to permit any conclusions concerning the upward migration of water from below, to, or through the salt layers or interest.

B.2.4 Tectonic Factors - Paradox Basin

Tectonic factors have been and are being addressed by an evaluation of remote-sensing data, field mapping, review of historical records, and microseismic monitoring.

Most of the faults in the basin are associated with the salt anticlines. Normal faults are found along the crest of these structures, where collapse resulted from the solution of the underlying salt. Pre-salt subsurface faults have not been traced into the post-salt surface faults.

The Paradox basin is a region of low seismicity historically. Out to 200 miles from its borders, 1,175 earthquakes of Richter magnitude 1.0 or greater have been identified to date. Of these, only 17 were in the basin proper, the largest being a magnitude 4.3 quake that occured in the extreme northwest corner of the basin near Green River, Utah, in July 1953 (21).

B.2.5 Resource Factors--Paradox Basin

The major known or potential energy and mineral resources in the basin are oil and gas, potash, and uranium/vanadium. Major deposits of salt are also present. Minor amounts of some base and precious metals have been produced (23). Possible conflicts with use of the land for a waste repository are being evaluated.

Portions of the core in Salt Valley, when removed from the core barrel, were bleeding very minor amounts of hydrocarbons. Only trace amounts of potash minerals were present in the core.

Hydrogen sulfide gas was detected at one of the 1,250-ft holes while drilling in caprock overlying the salt. The gas persisted only a short period of time. It was detected when a small amount of fluid, recovered during testing of the caprock, was exposed to atmospheric pressure.

B.3 Permian Basin

The Permian basin is a series of sedimentary basins in which rock salt and associated salts accumulated during Permian time more than 200 million years ago. It includes the western parts of Kansas, Oklahoma, and Texas, and the eastern parts of Colorado and New Mexico. Since Permian time the basin has been relatively stable tectonically, although some parts have tilted and warped, undergone periods of erosion, and been subjected to major incursions of the sea.

The Permian basin is one of the salt regions identified in a national screening as having potential for the siting of repositories (18). A regional characterization has been completed, and area-level studies are under way in the Palo Duro and Dalhart basins of Texas, described below in B.3.1.

In the New Mexico portion of the Permian basin, search for sites suitable for the Waste Isolation Pilot Plant led to the Delaware basin and the subsequent selection of a proposed site in the Los Medanos area which is discussed in B.3.2.

B.3.1 Palo Duro and Dalhart Basins

B.3.1.1 Summary

The Palo Duro and Dalhart basins (Figure B-5) were identified as areas with potential for siting a waste repository through a screening of the Permian Basin bedded-salt deposits (24). These basins are located largely in the Panhandle of Texas. The area is typified by an almost featureless plain

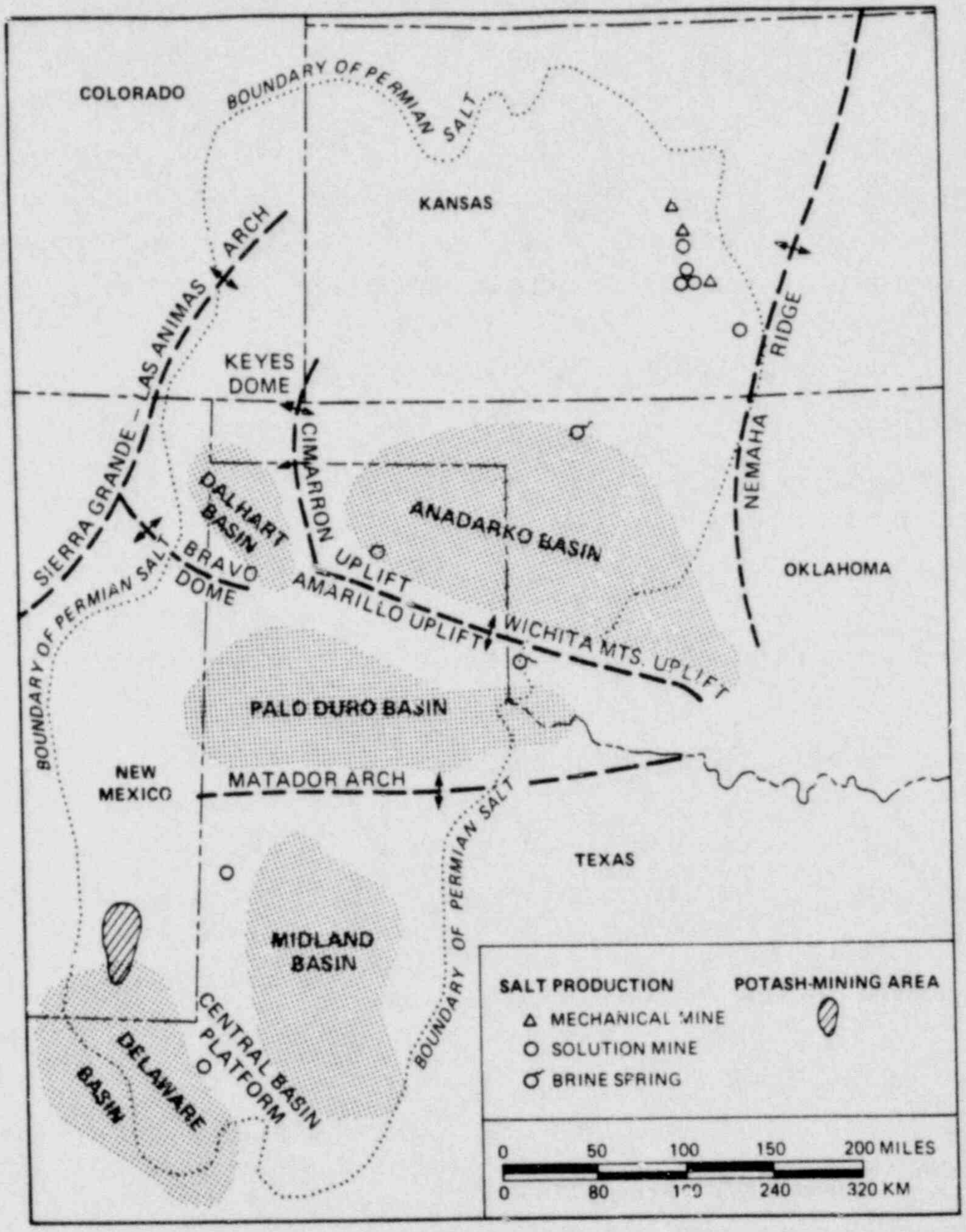


Figure B-5. Map of Permian Basin Salt Area in Southwestern United States Showing the Principal Tectonic Provinces.

Source: (Reference 18) K.S. Johnson and S. Gonzales, Salt Deposits in the U.S. and Regional Characteristics Important for Storage of Radioactive Waste, Y/OWI/Sub-741411, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

that is dissected by headward-cutting streams and is underlain by nearly horizontal Mesozoic and Cenozoic sedimentary formations.

Current investigation are in a subregional evaluation phase in which some field work has been done. Data available for direct analysis include (i) 8,000 ft of salt-bearing core, (ii) petroleum source-rock quality and thermal maturity data for resource assessment studies, (iii) drill-stem test data for regional hydrogeologic studies, and (iv) quantitative data on the climatic history, erosional potential, and shallow subsurface salt dissolution for predicting the long-term geomorphic integrity of the Texas Panhandle.

The data assembled to date are preliminary. Detailed area and site investigations began in FY 1980. Specific questions pertaining to hydrology, tectonics, geology, and resource evaluations will be the subjects of proposed investigations (25). However, based on the abundance of evaporite deposits which are interbedded with potential barriers to ground water migration (e.g. shale), the remoteness of the region, and its marginal potential for resources, the region is a likely candidate for further evaluation.

B.3.1.1 Geologic Factors--Palo Duro and Dalhart Basins

Upper Permian salt-bearing strata in the Texas Panhandle are composed of salt, anhydrite and/or gypsum, some limestone, and red beds. These rocks occur in cyclic units and were deposited in a range of shelf, supratidal, and terrestrial environments. Upper Permian rocks may be subdivided into genetic units that record regional changes in facies patterns (Figures B-6 and B-7).

Each genetic sequence can be subdivided into second-order cycles, which record more localized variations of shifting facies patterns. The Tubb Formation is the upper part of a major genetic unit comprising strata of the Lower Clear Fork and Tubb Formations. Lower Clear Fork strata lack the thick red-bed tongues exhibited by the Tubb Formation and record the early-stage dominance of coastal evaporite and carbonate environments in the study area. Tubb strata record the late-stage dominance of mud-flat environments, which migrated basinward from updip areas.

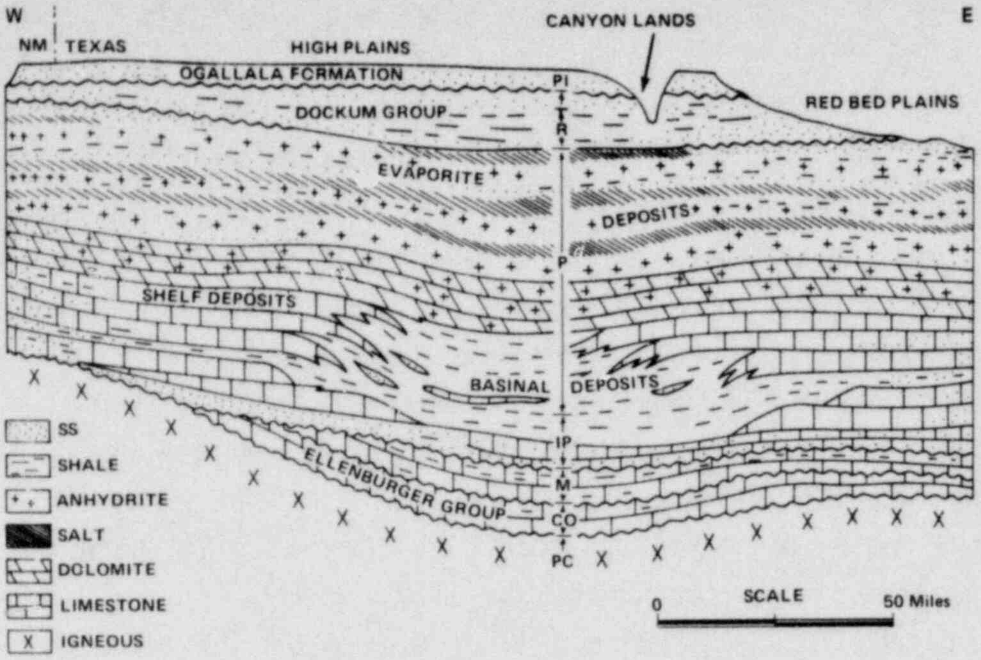


Figure B-6. Schematic East-West Section Across the Palo Duro Basin, Texas Panhandle.

Source: (Reference 26) S.P. Dutton et al., Geology and Geohydrology of the Palo Duro Basin, Texas Panhandle, pp. 87-95, the University of Texas at Austin, Bureau of Economic Geology Geological Circular 79-1, Austin, TX, 1979

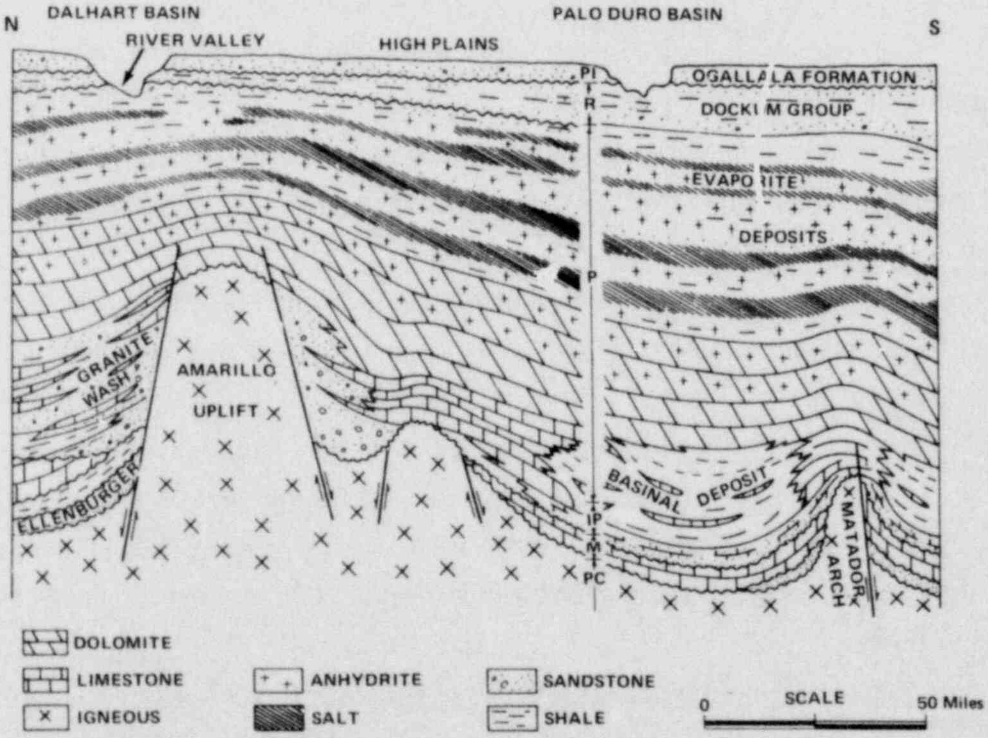


Figure B-7. Schematic North-South Section Across the Dalhart Basin, Amarillo Uplift, Palo Duro Basin, and Matador Arch, Texas Panhandle.

Source: (Reference 26) See above.

The Upper Clear Fork and Glorieta Formations comprise another major genetic unit and exhibit a facies succession similar to that of the Lower Clear Fork and Tubb rocks. The Upper Clear Fork and Glorieta strata contain a high proportion of siliciclastics intercalated with evaporites. Upper Clear Fork rocks record the early-stage dominance of coastal evaporite and carbonate environments in the study area. Glorieta rocks record the late-stage dominance of mud-flat environments. Continental terrestrial salt-flat environments also developed extensively during the deposition of the Glorieta Formation in landward areas of the northern Panhandle. Numerous second-order cycles displayed by Upper Clear Fork-Glorieta strata are delineated in the stratigraphic column (Figure B-8).

The San Andres Formation in the region was deposited mainly in coastal evaporite and carbonate environments. These strata lack significant quantities of the intercalated siliciclastics characteristic of the older Clear Fork strata; consequently, massive red-bed tongues deposited on mud flats are largely absent. Thick, laterally persistent, shallow-marine carbonate beds in the lower part of the San Andres record a marine transgression that was extensive over large portions of the study region.

Post-San Andres formations are composed dominantly of siliciclastics and salt deposited in evaporite, mud-flat, and terrestrial eolian/continental sabkha environments. Two major post-San Andres salt-bearing units are the Seven Rivers and the Salado Formations (26), in which mudstone-dominant beds interfinger basinward with more massive salt. Salt facies relationships are similar to those observed in the updip Clear Fork evaporite facies. Salt environments extended south and west of the Panhandle area into the Midland and Delaware Basins. In the Panhandle area, red beds intertongue downdip with the salt. Because of the extent of the generally flat-lying deposits, it is anticipated that the region can be adequately characterized. Additional information on the nature and the extent of facies changes throughout the study region is being compiled (25).

B.3.1.2 Hydrologic Factors--Palo Duro and Dalhart Basins

The Palo Duro and Dalhart basins are old, compacted basins that are marked by both shallow and deep-basin hydrologic systems (27). The thick

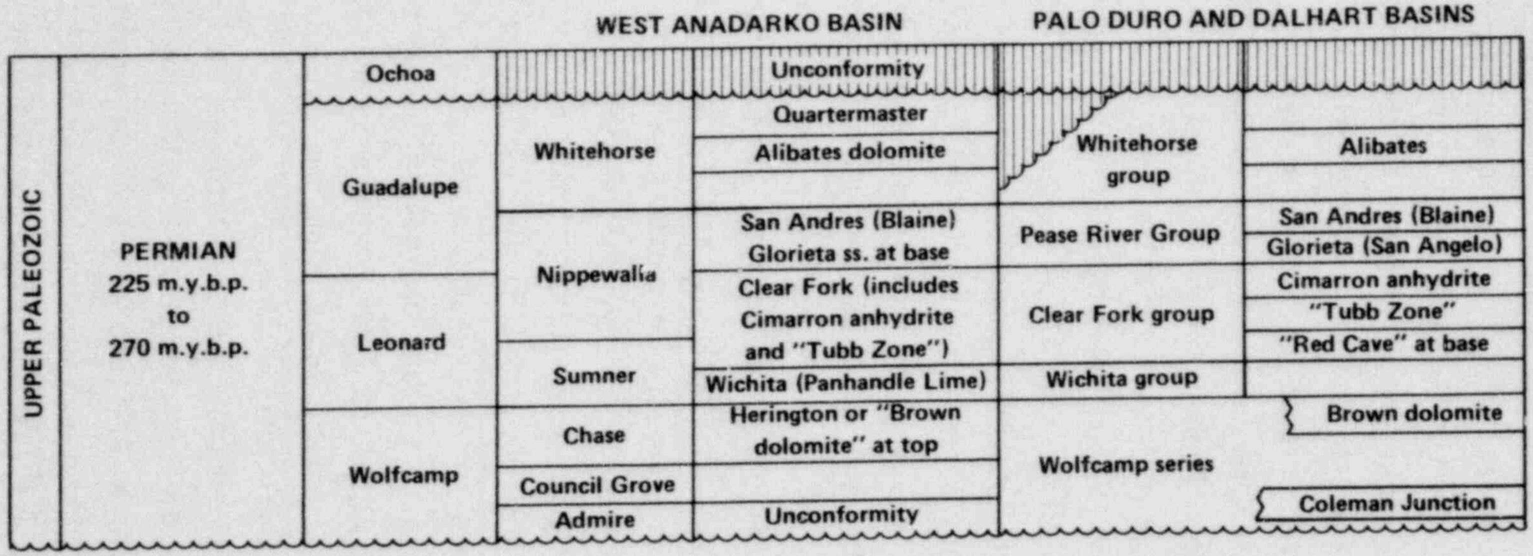


Figure B-8. Stratigraphic Correlation Chart, Texas
Panhandle Portion of the Permian Basin

Source: (Reference 26) S.P. Dutton et al., Geology and Geohydrology of the Palo Duro Basin, Texas Panhandle, pp. 87-95, the University of Texas at Austin, Bureau of Economic Geology Geological Circular 79-1, Austin, TX, 1979 sequence of alternating salt and clayey layers in the study area shows some potential for providing hydrologic barriers comprised of clayey interbeds.

Potentiometric-surface maps of Pennsylvanian and Lower Permian shelf and shelf-margin carbonate units reflect the primary directions of deep-basin ground water flow. In the Dalhart basin, recharge apparently occurs at the western side of the basin and discharge at the eastern side, near the northwestern end of the Amarillo uplift. In the Palo Duro basin, ground water in the Lower Permian aquifer tends to flow from west to east and northeast, discharging along the southeastern and northeastern edges of the basin. Regional flow in the deeper Pennsylvanian aquifer, however, is not marked by simple patterns, which may reflect a relative lack of data. In both aquifers (Lower Permian and Pennsylvanian) there is evidence that gradational changes (vertical and horizontal) in rock type control the distribution of the hydraulic head and hence the regional ground water flow.

The availability of hydrologic data for water-bearing units in the region varies directly with the amount of pumpage from the units. The best available data are for the major aquifer of the High Plains, the Tertiary Ogallala Formation. Limited data are available for the Cretaceous, Triassic, and Permian units below the Ogallala Formation because the water is generally saline and is rarely used for domestic and agricultural purposes. Small amounts of water are pumped from Permian aquifers of the High Plains.

There is a zone of active dissolution in Permian bedded salts which closely parallels the escarpment of the southern High Plains (24). If this relationship is not fortuitous, then the retreat of the escarpment and the southward and westward expansion of the dissolution zone may occur at similar rates. To ensure safe storage of radioactive waste, the integrity of the storage site must be protected from exposure to erosion or salt dissolution. Therefore, it is necessary to develop the capability to predict the rates of scarp retreat and salt dissolution.

The average annual solute discharge for the southern High Plains of Texas from 1969 to 1974 was 2.7572×10^6 tons of dissolved solids per year, including 1.1343×10^6 tons of chloride and 0.513×10^6 tons of sulfate (28). Nearly half this load was supplied by the Prairie Dog Town Fork of the Red River, the average load of which, for the same period, was equivalent to 119.54×10^5 ft^3 of halite. The sampling technique suggests that these are minimum values for solute loads, because discharge through alluvium is not included. An analysis of these salt-dissolution rates indicates that, at present, salt dissolution is more rapid along the eastern Caprock Escarpment than along the Canadian River Breaks. The maximum mean annual horizontal rates of salt dissolution range from 0.0019 to 0.1138 ft/yr. The maximum rate of vertical dissolution ranges from 0.767×10^{-5} ft/yr to 30.886×10^{-5} ft/yr (28).

Three time periods were used in analyzing the retreat of the eastern Caprock Escarpment: (i) since the end of deposition of the Ogallala Formation about 3 million years ago, (ii) since the end of deposition of the Seymour Gravel about 600,000 years ago, and (iii) since the deposition of a Holocene terrace about 8,000 years ago (24) (Table B-1). Although the rates of slope retreat probably varied with climatic cycles, determining slope retreats over long time intervals averages out variations.

The initial results of these studies are being reviewed. The representativeness of a single slope-retreat rate is always questionable, particularly if it is used in predicting future rates. However, since the slope-retreat rates calculated by three different methods and the maximum salt-dissolution rates established for areas east of the High Plains differ by less than a factor of 4, some credibility is given to these rates.

Table B-1. Time Periods: Retreat of Eastern Caprock Escarpment

Area	Time (years) Required to Retreat 1 km
Little Red River basin	7,200-8,600
Caprock Escarpment retreat since deposition of the Seymour Gravel	5,500
Caprock Escarpment retreat since deposition of the Ogallala Formation	9,000
Slope retreat of the Canadian River Valley near Fritch, Texas	24,000-32,000

Source: (Reference 24) T. C. Gustavson et al., Locating Field Confirmation Study Areas for Isolation of Nuclear Waste in the Texas Panhandle, ONWI/79/E-511-00300-4, Bureau of Economic Geology, University of Texas at Austin, Austin, TX, March, 1980

B.3.1.3 Tectonic Factors--Palo Duro and Dalhart Basins

Two northeast-trending fault systems occur along the western Caprock Escarpment in eastern New Mexico. The Bonita fault extends more than 10 miles. Another fault (referred to here as Alamosa fault) lies along Alamosa Creek and extends more than 7 miles. Both fault systems are grabens, consisting of normal faults dipping to the northwest and one or more antithetic faults dipping to the southeast.

The Alamosa fault system extends along Alamosa Creek about 7 miles. However, differences in elevation of stratigraphic units on either side of Alamosa Creek Valley and alignment of the creek along the fault suggest that the fault may extend southwestward for several kilometers beneath Quaternary alluvium. A stratigraphic cross section constructed across the lower reach of Alamosa Creek and across the extension of the Alamosa fault indicate that the San Andres Formation is stratigraphically continuous beneath the fault projection. The overlying Artesia Group (equivalent to the San Andres Formation in Texas) thins approximately 180 ft near the fault. Thinning of the Artesia Group is due to the dissolution of salt beds. Thinning of the Artesia Group has allowed collapse of the overlying strata, including the Alibates Dolomite Lentil. The Alamosa fault system cuts strata of both the Triassic Dockum Group and the Tertiary Ogallala Formation, indicating that the fault is at most late Tertiary and possibly Pleistocene in age. The Alamosa fault may be a southeasterly extension of the Bonita fault.

The Bonita fault displaces Permian, Triassic, and Cretaceous rocks, but Pliocene Ogallala sediments overlying the fault are not displaced. Therefore, it appears that the Bonita fault is significantly older than, and not directly related to, the Alamosa fault system. A stratigraphic section of the area between the two faults shows no evidence of abrupt structural displacement, although salt dissolution has occurred and increases northwestwardly. Analysis of a stratigraphic section across the fault, based on geophysical and lithologic well logs, suggests that approximately 250 ft of thinning occurs in the Artesia Group, which can be explained by dissolution of salts and the subsequent collapse of overlying strata, including the Alibates. The subsurface collapse is manifested at the surface by the Bonita fault system, where surface displacement is also approximately 250 ft.

Small-scale faulting has occurred in Hall County, south of the Prairie Dog Town Fork of the Red River and approximately 12 miles west of Estelline, Texas. Six fractures were recognized by freshly repaired cracks in a paved secondary road. Faults trending across the road were continuous, with open fractures to 4 in in adjacent cultivated fields. Vertical displacement across faults on the road surface ranged from 0.4 to 1.6 in. All the faults were aligned between N 25° E and N 60° E. These faults are associated with two large, undrained depressions. This area of Hall County lies within the zone of active salt dissolution, and Estelline Spring, the largest saline spring in this region, is only 13 miles east of the fault area. Undrained depressions, a saline spring, and faults suggest that salt dissolution and collapse are occurring beneath the fault area (26).

B.3.1.4 Resource Factors--Palo Duro and Dalhart Basins

The potential for petroleum resources in the basins is being evaluated by studies of source rock quality and thermal maturity. Current data suggest only marginal potential for petroleum resources. Further studies are proposed to thoroughly evaluate petroleum and mineral resources (25).

Source rock quality is measured by the total organic carbon content (TOC). To determine whether sediments in the Palo Duro basin contain sufficient organic matter to generate hydrocarbons, 341 samples collected from 20 geographically scattered wells were analyzed for TOC. Samples were taken from a range of depths and stratigraphic intervals, with sampling concentrated in Pennsylvanian and Lower Permian shales from basin and prodelta facies.

The total organic carbon content ranged between 0.008% and 6.866%. One hundred thirty-four samples contained more than 0.5% TOC, which is the cutoff between poor and fair hydrocarbon source rocks. The highest values of TOC occur in Upper Permian San Andres dolomite in the southern portion of the basin. Pennsylvanian and Wolfcampian basinal shales contain up to 2.4% TOC and are fair to very good source rocks. Organic carbon in Pennsylvanian and Wolfcampian strata is most abundant in the basin-center deposits (26).

Source beds in the Palo Duro basin had to reach sufficiently high temperatures to generate hydrocarbons from disseminated organic matter.

The physical characteristics of the remaining organic material, especially color and reflectance, indicate maximum paleotemperatures. Kerogen color and vitrinite reflectance were studied for all samples containing more than 0.5% TOC (26).

B.3.2 Los Medanos Site

B.3.2.1 Summary

Geologic and hydrologic investigations of a site have been under way in southeastern New Mexico since 1972. This site is within a region of the Delaware Basin and is about 26 miles east of Carlsbad, New Mexico (Figure B-9). This site is well characterized and has previously been proposed by the Department for the Waste Isolation Pilot Plant (WIPP), a facility which was authorized by Congress as a site for the disposal of transuranic wastes from the defense program. The site investigations have been summarized in a comprehensive Geological Characterization Report (29).

The data collected to date indicate that the site has all the desired features, with the possible exception of some conflict with natural resources. While it is possible that future exploration at depth or improved understanding of geologic processes could reveal aspects undesirable for a repository, these prospects are unlikely. Calculations performed for a potential breach of a WIPP repository by future human activity, believed to be more likely than natural failures of the geologic barriers, indicate that, even under extremely conservative assumptions, the potential hazard to the general population is very slight--less than that from naturally occurring radiation (30).

B.3.2.2 Geologic Factors--Los Medanos

The geologic investigations at the Los Medanos site include field mapping at the site and in adjacent areas of interest; the drilling of more than 70 new holes, which included taking over 12,000 ft of core; studies of the petrologic, mineralogic, geochemical, and thermophysical properties of core (and fluid) samples; releveling 200 km first-order vertical control

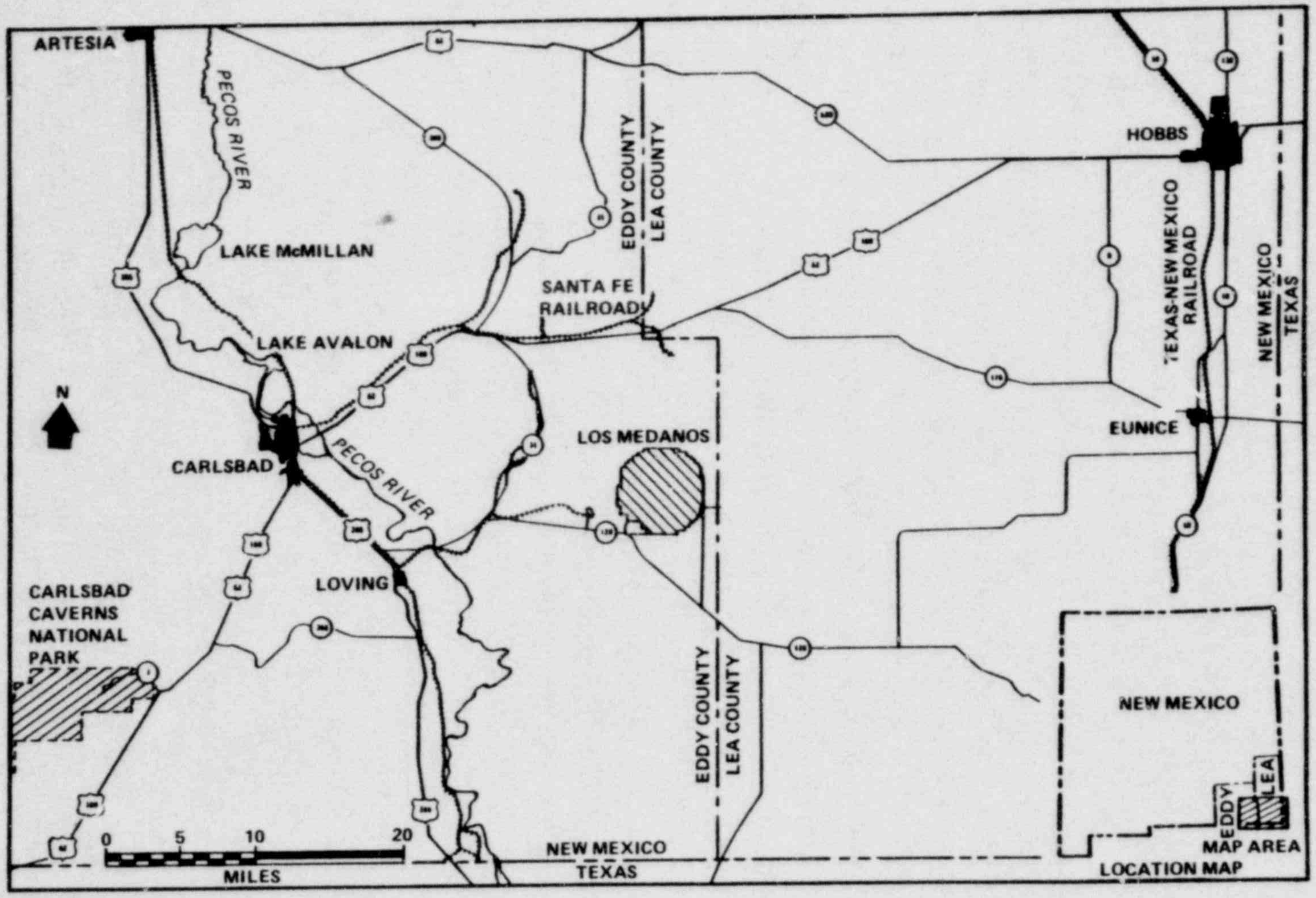


Figure B-9. Location Map for the WIPP Site in the Los Medanos Area lines; and establishing 300 km of new first-order vertical control lines for examining subsidence and tectonic activity.

A normal stratigraphic sequence of evaporite beds (Castile, Salado, and Rustler Formations) has been established as existing at the site (Figure B-10). The evaporites have a thickness of about 3,500 ft with top and bottom depths of about 500 to 4,000 ft respectively. Within the site there is a gentle dip of about 1 degree eastward which may have slight undulations imposed upon it. The lower evaporites in the northern part of the buffer zone show structural (not necessarily tectonic) complexities which are still under study. Mechanical flow of halite has been observed in some cores from the area and contributes some structural complexity. The evaporite section is not pure halite, and several minerals have significant sorptive properties. In flowthrough tests, sorption ratios for nonchelated ions (particularly fission products) correspond very well with those from batch measurements. For actinides, the systematics of sorption are considerably less well defined. Distribution coefficient (K_d) values of less than 1 are indicated for technetium and iodine in different rock types; the K_d values for other fission products generally range from 1 to 10^3, while those for the actinides range to about 10^4. High K_d values are more common in the dolomitic aquifers of the Rustler (30). Thermomechanical testing of evaporite rocks, particularly rock salt, at the Los Medanos site demonstrates the similarity in properties to evaporites elsewhere. Thermal conductivity averages about 5.75 W/m-K for rock salt in the lower Salado. Mechanical testing (with confining pressures of less than 3000 psi, a deviatoric stress of less than 5,000 psi, and temperatures of up to 250^{0}C) has been conducted at strain rates of 10^{-6} sec^{-1} (quasistatic) to 10^{-10} sec^{-1}. In situ testing of stress in boreholes has not been undertaken at the site.

B.3.2.3 Hydrologic Factors--Los Medanos

Hydrologic evaluations of the Los Medanos site have used both existing and new borehole data for aquifers above and below the repository horizon. To date, 45 new hydrologic boreholes at 24 different locations have been drilled. Two of these holes, one north and one south of the site, sample the Delaware Mountain Group aquifer, which is below the salt beds. The

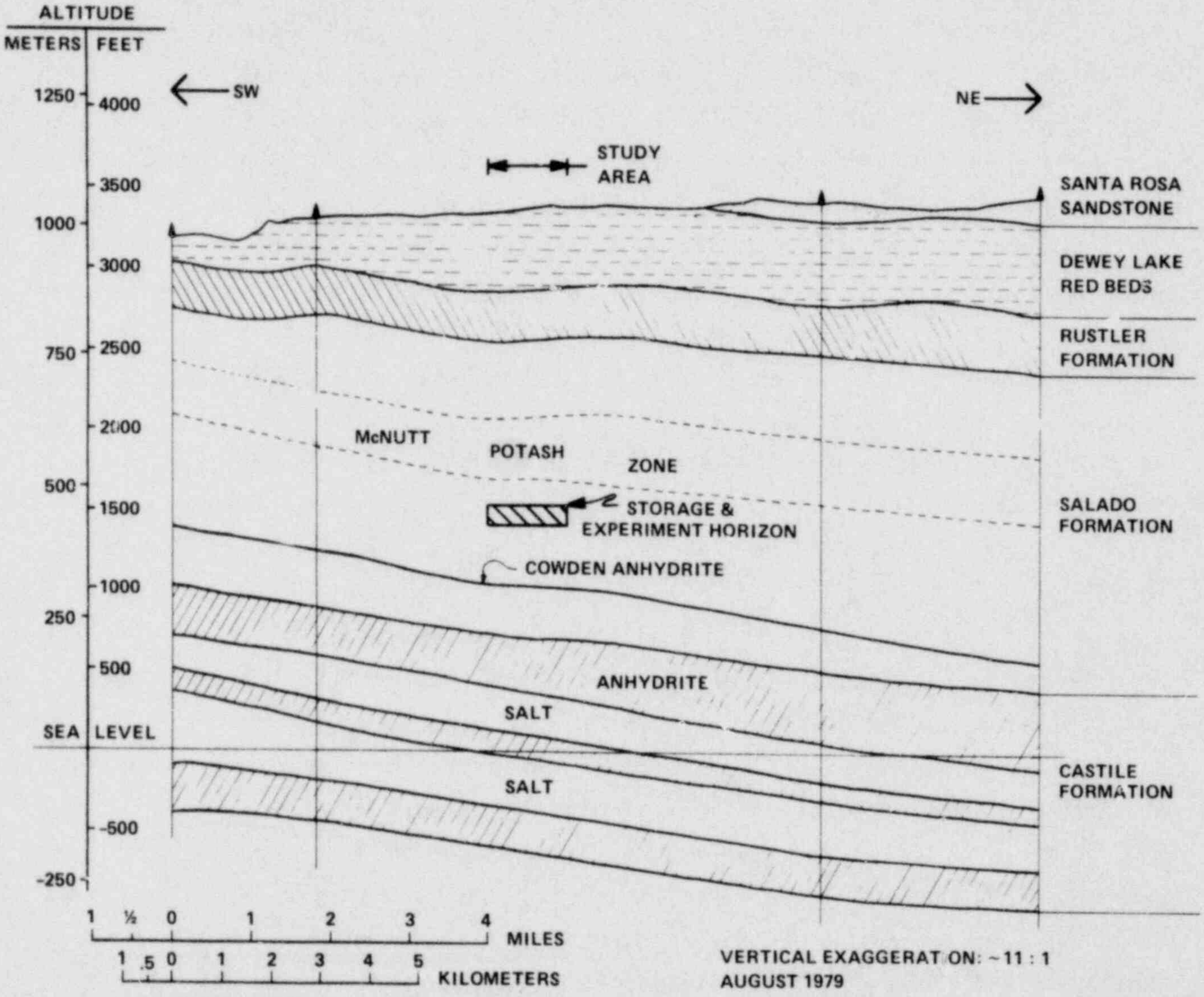


Figure B-10. Stratigraphy at Los Medanos regional hydrology of these deep formations has been characterized in previous investigations, and the new holes have confirmed the earlier results (31).

The objectives of the hydrologic testing program are to determine the static head or reservoir pressure, the water-yielding and transmitting potential of the rock strata, and the chemistry of formation water. Tracer studies are now under way to quantify the pertinent parameters and their degree of variability in the Culebra and Magenta aquifers. Special attention will be given to fracture permeability. These data are then used in development of a model of the local and regional hydrologic system for evaluating the long-term aspects of salt dissolution and, in the event of a repository breach, the transport of radionuclides to the biosphere. The method used in developing the model is to characterize the hydrologic system in detail at the repository and to acquire data at successively larger intervals as the distance from the repository increases. There are four triangular nests of three hydrologic holes each within the repository site. These nests have spacings of 50 to 100 ft between holes. The next larger scale of data acquisition employs spacings of 0.5 mile between test points. These holes in turn form part of a still larger grid spacing of hydrologic holes that surround the site at separations of 2 or more miles and which provide data on regional hydrology.

Since the most plausible natural mechanism for breaching the repository is the dissolution of the salt barrier by ground water, this aspect was studied extensively. Regional geologic studies (28, 32) in the Delaware basin have revealed areas of past and present dissolution activity. Knowledge of these regional dissolution fronts and of local collapse features induced by salt dissolution prompted geophysical investigations to ensure that such features do not exist in, or near enough to, the site to present a hazard to the long-term integrity of the repository. Geologic studies (32, 33) have indicated that the regional dissolution front west of the site area is progressing eastward at a rate of less than 6 to 8 miles per million years. Conservatively determined values indicate that the repository beds will not be breached by regional dissolution for many millions of years. Since these are average rates covering the past 600,000 years, the effect of previous pluvial cycles is included and, consequently, the forecast also incorporates the effect of similar future pluvial cycles. More recent studies (34) have shown the above dissolution rates to be very conservative since the observed dissolution is now known to have occurred over a much longer time. The mapping also shows that no development of known breccia pipes has occurred in the region in the past 500,000 years.

The parameters of the hydrologic system at the site are quite favorable with respect to the transport of radionuclides. While the actual heads indicate that water would not carry isotopes up into the Rustler aquifers, this assumption has been made to permit conservative safety analyses. The ground water gradient at the site indicates that water in the Rustler aquifers would discharge to the Pecos River at Malaga Bend. The shortest possible path length is 15 miles, and the transit time for water is more than 1,000 years. Studies under way will allow a more precise estimation of this time. Aquifers above the repository produce less than 1 gpm and have such a high dissolved-solids content that the water is not potable. Conventional dating techniques have not been useful because of the geochemical nature of the fluids and the aquifer rocks.

The following conclusions on hydrology can be drawn from studies by the U.S. Geological Survey (35):

1. The water levels of fluid-bearing zones in the Rustler Formation show that the hydraulic potential decreases with depth, indicating downward fluid movement in rocks above the salt should there be any openings. However, the potential-head differences between fluid-bearing units indicate no vertical hydraulic connection.

2. The distribution of head in the Culebra Dolomite indicates ground water flow southeast across the site and then south-southwest, with the gradient varying from 7 to 120 ft/mile. Transmissivity varies from 140 ft^2/day on the flanks of Nash draw to 10^{-1} ft^2/day near the center of the site and 10^{-4} ft^2/day on the east side. This variation is attributed to the dissolution of salt in the Rustler, which decreases from the complete removal of salt in the west to little or no removal in the east.

3. Potential-head measurements in the Magenta Dolomite indicate fluid movement to the southwest. The hydraulic gradient is 50 ft/mile, and transmissivities range from 0.01 to 2.0 ft^2/day.

4. Fluids in the Culebra and Magenta Dolomites apparently move primarily along fracture systems and through low-yielding fractured rocks.

5. Very low yields of brines were found along the Rustler-Salado contact, with transmissivities ranging from 10^{-1} to 10^{-5} ft^2/day.

6. Preliminary evaluation of tests on Bell Canyon sands at drill hole AEC-8 shows that the potentiometric surface, corrected to fresh water density, is higher than similarly corrected levels of fluid zones in the Rustler.

7. Preliminary data from drill holes outside the Los Medanos site indicate a ground water boundary at a surface ridge between the site and San Simon Swale.

8. The ground water gradient for the Santa Rose Sandstones appears to be determined by a hydrologic divide west of San Simon Swale and by local pumping practices near the Pecos that cause flow into the Pecos rather than directly into Texas.

B.3.2.4 Tectonic Factors--Los Medanos

Investigations of structural features and tectonic processes involve seismic reflection methods (over 1,500 miles of industry data, 152 line miles of new data), resistivity (over 9,000 measurements), gravity, aeromagnetics, boreholes and borehole geophysical logging, seismological monitoring, mapping, and radiometric dating (36).

The geologic record demonstrates large-scale downwarping in Paleozoic time resulting in a thick sedimentary sequence. Since Paleozoic time, the area has been emergent with a marine transgression in the Cretaceous. Deposition of the Ogallala Formation and the formation and preservation of the Mescalero caliche (500,000 years old) are indicators of the general tectonic and climatic stability of the Los Medanos area. Level lines in the area indicate uplift relative to the site area at the Guadalupe Mountains and the Diablo Plateau. Releveling of elevation lines will help establish the present rates of uplift or subsidence in the basin. No faults of tectonic origin are known in the Mescalero caliche at or in the vicinity of the Los Medanos site. The nearest known recent fault is west of the Guadalupe Mountains, about 65 miles west-southwest of the site. Three seismic events have been reported within 35 miles of the site; the magnitudes ranged from 2.3 to 3.6 (37). Seismic activity is common at slightly greater distances at the Central Basin Platform, but studies of the activity indicate water injection for secondary recovery from oil fields as the probable cause (38). The nearest igneous activity is a dike that intrudes the evaporites about 9 miles from the site. Potassium/ argon dating of the dike indicates an age of about 35 million years. Larger intrusives of this age are known 75 miles west-southwest of the site; further west there are late Tertiary extrusives.

B.3.2.5 Resource Factors--Los Medanos

Natural resources within any major salt basin are an ever-present potential. Some potash and potentially some hydrocarbons exist within the buffer zones established for the WIPP repository. The estimated amounts within zones I, II and III (which may not be exploited if the Los Medanos site is developed) are 13.3×10^6 tons of langbeinite, 21.1×10^9 ft^3 of gas, and 258,000 bbl of condensate. The potash resources have been established by direct drilling and core assay; the hydrocarbon reserves have been statistically estimated utilizing seismic information on geologic structure and production experience in the region.

Natural gas reserves, since they occur at depths of 10,000 to 15,000 ft, well the below proposed repository horizons, may be produced by deviated drilling without breaching the salt beds inside zones I, II and III. It may also be possible to extract the potash from above the repository without affecting long-term safety, but this prospect has not yet been firmly established.

The consequences of future human intrusion by exploratory drilling into the repository have been evaluated for the WIPP; the results show that the consequences would not pose significant hazards to mankind in the future (38). In addition, the water in aquifers at and near the Los Medanos site is not available in quantities and quality that make its use ...ely for domestic or irrigation purposes.

B.4 Salina Basin

B.4.1 Summary

The Salina basin as here defined is a portion of the northeastern United States underlain by bedded salt in a rock unit called the Salina Group or Formation of Late Silurian age. This salt formation is present in the northern Appalachian and Michigan basins in parts of Michigan, Ohio, Pennsylvania, and New York (Figure B-11).

A regional study of the Salina basin has been made from the existing geologic literature and geologic data available from public and private sources. For the Ohio and New York portions of the Salina basin, the regional studies have been completed (39, 40) and initial evaluations have identified study areas that appear to be geologically favorable for more detailed field investigations. The Michigan portion of the Salina basin has not been studied in sufficient detail to allow the identification of study areas. However, it is known that Michigan has salt beds of sufficient thickness and extent to meet present specifications for waste repositories and that these beds occur at suitable depths.

No field investigations have been carried out by the Department in the Salina basin. Some field work in support of repository siting has been done by the U.S. Geological Survey in New York and in Pennsylvania.

The Salina basin is considered to be tectonically stable; it has low seismicity and is far from crustal plate boundaries. The potential for uplift or subsidence in the next million years appears to be minor. Even at extreme rates of denudation, surface lowering by stream erosion will not threaten a repository located at least 1,000 ft below the surface. Although glacial scour in the Finger Lakes region of New York has reached estimated depths of 1,200 ft to possibly as much as 1,800 ft, this process is not general and was caused by specific geologic conditions. The amount of glacial scour in valley areas needs to be investigated further, but upland areas appear to be favorable with regard to potential future glacial erosion.

New York salt beds were deformed by Paleozoic thrust faulting, folding, and probable decollement with tear faults. The structure is complex,

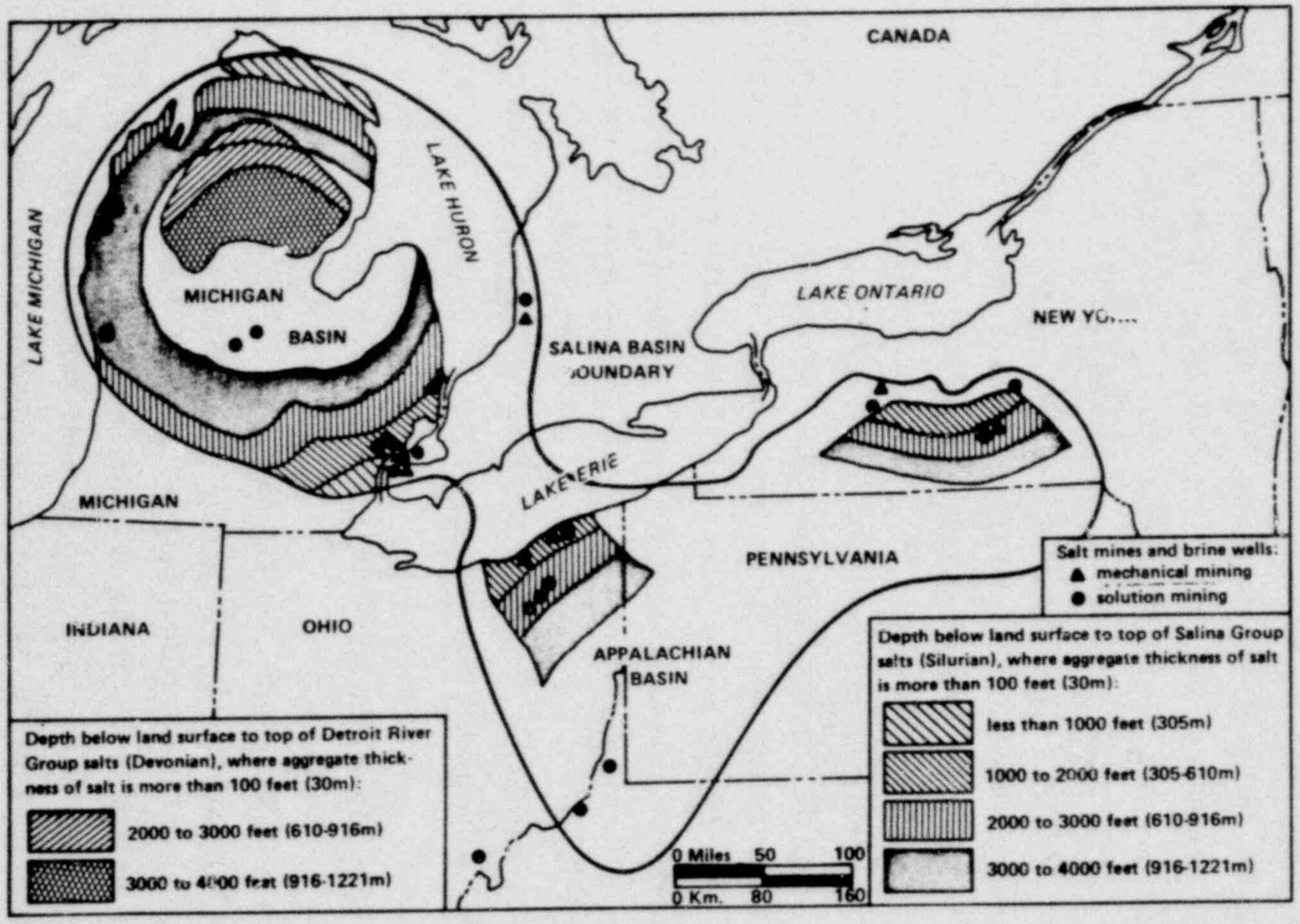


Figure B-11. Michigan and Northern Appalachian Basins (Salina Basin)

Source: (Reference 18) K.S. Johnson and S. Gonzales, Salt Deposits in the U.S. and Regional Characteristics Important for Storage of Radioactive Waste, Y/OWI/Sub-741411, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

but present information indicates that large areas within the State should be investigated further.

A large area in Northeastern Ohio meets preliminary screening requirements: salt with an aggregate thickness of 75 ft or more, and depths between 1,000 and 3,000 ft. Ohio has more oil and gas production than New York and a high potential for future resource development. Resource conflicts may be severe for siting a repository in Ohio.

Regional analyses suggest that Michigan is geologically favorable. No detailed screening has been done, but the stratigraphy and the structure of the salt beds appear to be promising.

Much additional information is needed before a repository site could be identified in the Salina basin. The deep ground water circulation and flow systems are not known. The detailed composition of the salt beds is not known. The water content and the mineral impurities of the salt need to be determined. In addition, the nature of facies changes within the salt beds and surrounding rocks needs to be evaluated. Dissolution of salt is probable in New York where the Salina Group crops out. The rates and processes of dissolution are not known and need to be evaluated. Northeastern Ohio does not appear to have significant present-day dissolution. The potential for oil and gas development in the Salina basin needs to be evaluated.

The regional geologic studies of the basin indicate that New York, Ohio, and Michigan all have both favorable and unfavorable geologic characteristics for a repository. At the present, no part of the basin can be judged acceptable or unacceptable for repository siting.

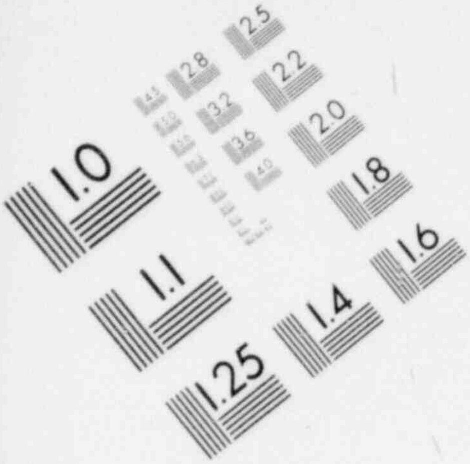
B.4.2 Geologic Factors--Salina Basin

The geologic factors addressed in the Salina basin studies were focused on the stratigraphic distribution and structure of the salt beds, the Salina Group and the enclosing sedimentary rock section (Figure B-12). The geologic history of the basin, the deposition sequences and source of sediments, the surficial geology, erosion and denudation rates, glacial erosion, the location of faults, the definition of joint patterns, and dissolution structure were also assessed. Most of the study effort necessarily concentrated on the geometric configuration of the salt beds and their structural

PERIOD	STAGE	MICHIGAN		EAST OHIO		WEST NEW YORK	
		Unconformity		Unconformity		Unconformity	
Silurian (400 m.y. b.p.)		Bass Islands		Bass Islands		Bertie - Akron	
	CAYUGAN	Salina Group	UNIT G	Salina Group	UNIT G	Salina Group	Camillus
			UNIT F		UNIT F		Syracuse
			UNITS A - E		UNITS A - E		Vernon
	NIAGARAN	Niagara Group	LOCKPORT	LOCKPORT		LOCKPORT	
				Clin-ton Grp.		Clin-ton Grp.	
	ALEXANDRIAN	Cata-ract Group		Albion Group		Medina Group	

Figure B-12. Stratigraphy of the Salina Salt Group

Source: (Reference 39) Stone and Webster Engineering Corp., Regional Geology of the Salina Basin, Battelle/ONWI/Sub-E512-06001, (two volumes), Battelle Memorial Institute, Columbis, OH, 1978



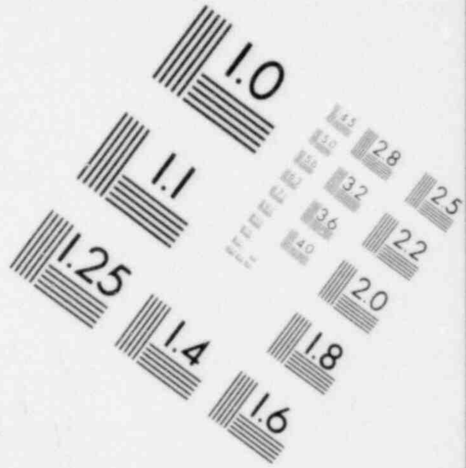
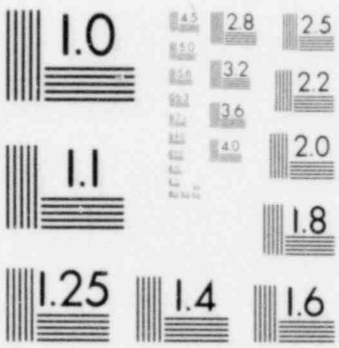
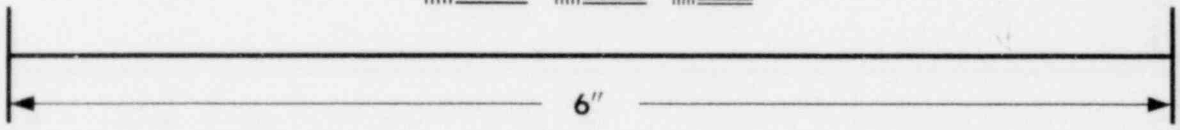


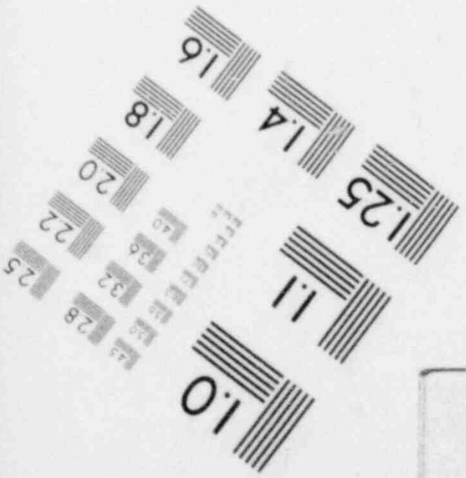
IMAGE EVALUATION
TEST TARGET (MT-3)

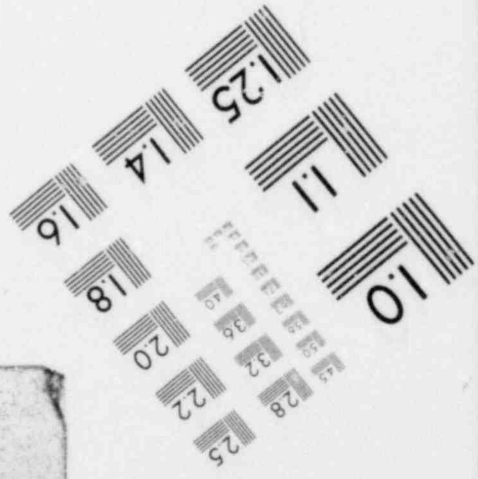




6″

MICROCOPY RESOLUTION TEST CHART





PERIOD	STAGE	MICHIGAN		EAST OHIO		WEST NEW YORK	
		Unconformity		Unconformity			
Silurian (400 m.y. b.p.)	CAYUGAN	Bass Islands		Bass Islands		Bertie - Akron	
		Salina Group	UNIT G	Salina Group	UNIT G	Salina Group	Camillus
			UNIT F		UNIT F		Syracuse
			UNITS A - E		UNITS A - E		Vernon
	NIAGARAN	Niagara Group	LOCKPORT	LOCKPORT		LOCKPORT	
				Clin-ton Grp.		Clin-ton Grp.	
	ALEXANDRIAN	Cata-ract Group		Albion Group		Medina Group	

Figure B-12. Stratigraphy of the Salina Salt Group

Source: (Reference 39) Stone and Webster Engineering Corp., Regional Geology of the Salina Basin, Battelle/ONWI/Sub-E512-06001, (two volumes), Battelle Memorial Institute, Columbis, OH, 1978 setting. The thickness, depth, and lateral extent of salt were the primary considerations in screening the region for areas suitable for further study.

The bedded-salt deposits of the Northeastern United States are confined primarily to the Michigan and Appalachian basins, which together constitute the Salina basin. The salt deposits are thickest and most extensive in the Michigan basin. Extensive but significantly thinner salt deposits accumulated in the Appalachian basin, with deposition centers in northeastern Ohio and north-central New York.

The evaporite deposits overlie Niagara dolostone throughout the region. The Salina Group in the Appalachian basin and the Salina and Bass Islands Groups in the Michigan basin contain Late Silurian salt beds interlayered with anhydrite, dolostone, and shale, which could provide multiple barriers to radionuclide migration in the ground water system. The total thickness of the upper carbonate and evaporite sequence is about 6,000 ft in the Michigan basin and about 3,000 ft in the Appalachian basin. Subsurface correlations with geophysical logs indicate that the thickness and the lateral extent of salt beds is different in each of the major centers of deposition, but the stratigraphic sequence is regionally persistent and therefore more readily characterized and modeled.

In New York, subsurface correlations show that salt beds occur almost entirely in the Syracuse Formation. The major salt-bearing unit is a thick sequence of evaporites with thin beds of dolomite and anhydritic shale. The evaporites are mainly salt with lesser amounts of dolomite and anhydrite. In New York, salt beds in portions of the Syracuse Formation are most promising for repository siting because of their thickness (combined sections up to 900 ft thick) and widespread distribution.

Unconformities in the stratigraphic section account for the absence of stratigraphic units in parts of New York. These unconformities control the distribution of the Oriskany Sandstone, an important aquifer. The hydrogeologic importance of the unconformity surfaces and the effect of the unconformities on the stratigraphic section above the Salina is not clear.

The most extensive and thickest salt beds in Ohio occur in the "F" unit of the Salina Group. A portion of the unit at the base has at least two salt beds that are more than 20 ft thick. In Geauga and Cuyahoga Counties, this unit contains as much as 100 ft of salt. Strata overlying this lower unit contain thick beds of halite separated by thinner beds of dolomite and anhydrite. The remainder of the "F" unit is either not a major salt-bearing interval or only a local salt-bearing interval.

Salt beds in Ohio appear to be much more persistent internally than those in New York. Some thickness anomalies do occur, which may be due to local stratigraphic control or post-depositional processes. However, these anomalies do not appear to reflect recent dissolution. No evidence has been found of naturally occurring salt dissolution anywhere near the area that has been considered in Ohio. In Ohio, unconformities of Early Devonian age occur in rocks above the Salina Group. These may have a bearing on the hydrology and distribution of aquifers above the salt.

In Michigan, salt beds occur in many units of the Salina Group and in the Detroit River Group of Devonian age. Regional information indicates that sufficient thicknesses of high-quality salt occur at depths between 1,000 and 3,000 ft beneath large areas in the Lower Peninsula of Michigan. The geologic structure in Michigan is subdued. Gentle folds and relatively small displacement faults appear to be related to basin subsidence, a very slow geologic process.

During most of the past 225 million years, the Salina basin region has been slowly denuded by stream erosion. Continued denudation could produce between 12 and 50 ft of surface lowering over a period of 250,000 years. The acceleration of stream erosion by orogenic deformation is considered unlikely because of the stable tectonic history of the region, the distance of the region from active plate margins, and the time required for tectonic events. Significant epeirogenic crustal uplift is also considered unlikely, but even at the most extreme rates of upwarping, maximum incisement by rivers would probably be less than 500 ft.

The most profound impact on the land surface of the study region was caused by glaciation during the last 3 million years. Glaciers have tremendous erosional capacity and, unlike streams, are not restricted by base levels. The maximum depth of glacial erosion in the Salina basin locally exceeds 1,200 ft. The depth of potential glacial scour in the future is considered a critical parameter in site selection. In New York the Finger Lake troughs, the most remarkable landforms of the study area, are deep valleys formed by glacial erosion in preexisting stream valleys. In the Seneca

Valley, glacial scour may be nearly 1,800 ft. Deep glacial erosion is suspected in the other Finger Lake valleys and in the valleys of the Cohocton, Canisteo, and Tioughnioga Rivers.

Throughout the New York study area, the Salina Group rocks are more than 1,000 ft deep. Geologic containments beneath the intervalley uplands are considered safe from erosional breaching. Available data indicate a potential breaching of the Salina rocks by future glacial scour along the Seneca Valley. Bedrock depths will have to be further investigated before judgments can be made about the other major glaciated valleys of the study area.

To the west in Ohio, relief is considerably more subdued than in New York. The buried bedrock surface represents a mature terrain, only slightly modified by glacial erosion. Glacial smoothing of the landscape was accomplished by burial rather than by planation.

The only valley in the Ohio area known to have had significant glacial deepening is the valley of the Cuyahoga River, where glacial scour may have reached a depth of 1,100 ft.

Throughout the Ohio area, the top of the Salina Group is more than 1,000 ft deep. Based on available data, it is considered probable that future glacial scour would not breach the Salina salt beds anywhere in the Ohio study area. However, further investigation is needed to verify this assumption.

B.4.3 Hydrologic Factors--Salina Basin

Hydrologic studies have shown that there are two systems of ground water flow in the Salina basin: (1) near-surface fresh water aquifers and (2) a variable-density system of saline aquifers 500 to 1,000 ft deep. Some information is available for the shallow fresh water zone, but the deep and interaquifer ground water flow patterns are generally unknown. Detailed data would be needed on both aquifer and fluid characteristics to determine the direction and the velocity of the deeper ground water flow, which would provide a basis for evaluating the isolation potential.

In New York, the depth to saline ground water varies from over 1,000 ft below the Appalachian plateau to less than 500 ft in the Ontario lowlands. The water below these depths is highly mineralized. The Salina group yields the poorest quality ground water.

In Ohio, principal bedrock aquifers are sandstones, shales, and conglomerates of Mississippian and Pennsylvanian age. As in New York, sand and gravel outwash deposits in river valleys are the most important aquifers and may yield over 1,000 gpm to a single well. The potable-water zone is thin, and the depth to saline ground water is about 500 ft or less.

In Michigan, principal bedrock aquifers are sandstones, limestones, and dolomites of Mississippian and Pennsylvanian age. The base of fresh water lies 200 to 400 ft below the surface, except in the northern portion, where it ranges up to 900 ft. Fresh water in bedrock aquifers is generally confined to areas where bedrock is overlain by permeable glacial deposits. Salt springs, mineralized lake and river waters, and brecciated zones indicate that, in some locations, the Salina salt is undergoing dissolution. Another indication is the absence of salt in zones where salt would be expected to be present.

If the ground water circulating in contact with salt deposits reveives little recharge or is the entrapped water of deposition, its quality will be that of a brine. The brine will be at or near saturation with salt, and the rate of dissolution will approach zero. This situation exists in most of the Salina basin because the salt is overlain by hundreds to thousands of feet of more recent geologic deposits containing water that is highly saline. The existence of such extensive salt deposits in the basin demonstrates that access by fresh ground water has been, and is, extremely limited. However, where salt crops out or is shallow and overlain by permeable materials, dissolution may be rapid.

Changes in the hydrologic regime that could be brought about by environmental changes have been addressed in a preliminary way. Important considerations would be glaciation, permafrost, and other extreme climate changes as well as mining and other man-induced stresses. These possible changes would have to be analyzed when the present-day conditions are more fully defined.

B.4.4 Tectonic Factors--Salina Basin

The Salina basin is in the Central Stable Region, an interior platform that has been tectonically stable for more than 600 million years. Tectonic events marginal to the Stable Region have affected the platform's rocks to some degree and have controlled the depositional patterns of the Appalachian basin to a large extent. However, the geologic structures that have developed in the Salina basin are limited to broad gentle flexures and minor folds and faults, indicative of mild tectonism.

The Paleozoic platform sediments overlie Precambrian basement rocks believed to be continuous with those exposed in the Canadian shield to the north. These basement rocks evolved over a long period of time through a series of orogenic, metamorphic, and igneous events. By late Precambrian, time these rocks had become stable, and the Precambrian tectonic trends are now inactive relict structures.

Tectonism during the Allegheny Orogeny affected the rocks of the Appalachian basin. The faulting associated with Alleghenian movements displaced Salina and overlying stratigraphic units northward and westward and caused local thickness changes and structural complexities. The effects are strongest in the New York part of the Appalachian basin and less so in the Ohio area. North-south faults in the New York part of the basin are notable because of their potential effect on repository siting. The Clarendon-Linden fault zone, near Attica, New York, is considered to be a seismically active, reactivated basement structure. Another fault, in the Seneca Lake area, shows a right lateral offset of hundreds of feet, cuts salt beds, and has surface emanations of brine; however, it is not associated with seismicity. Information on its association with the basement structure is yet to be developed.

Although there are a variety of faults in the New York study area, two types seem to be of major importance: (i) thrust faults within the upper salt beds and overlying strata and (ii) strike-slip faults that separate large detachment blocks and terminate downward at the thrust surface. The Clarendon-Linden fault zone does not affect the study area directly, but it projects to within a few miles of the study area. It is considered to be seismically active and may control the seismic design of any repository within the study area.

Thrust faults in the study area occur in the upper salt beds and cause overthickened salt strata. They are related to Alleghenian folds and are possible sources of ground water seepage in the salt horizons. Strike-slip faults are believed to occur in general northerly trends along the west side of Seneca Lake south to Pennsylvania and from west of Ithaca south into Pennsylvania. The extent and character of thrust faults and strike-slip faults in the New York study area need to be verified.

The Central Stable Region as a whole has been subject to gentle uplift since Paleozoic time. No major tectonic structures of regional extent are considered active or potentially active. Two areas within the Stable Region, the Clarendon-Linden trend and the Anna, Ohio, area have anomalous seismic activity associated with fault structure. Both of these zones appear to be reactivated basement faults with relatively small displacements.

Geologic structures in the Ohio study area are much less common than in the New York area. High-angle faults are found but are not common. No thrusts or strike-slip faults like those in New York are known. The faults in Ohio appear to be relatively local and of small vertical displacement. No faulting of regional extent is known. The faults may have some effect on the salt beds if they persist to the depth of the Salina. These faults need to be investigated to determine their configuration and origin.

In New York there are some indications from mine workings in the Salina salt that local stress levels in the rock may be sufficient to cause stability problems at the depths which would be required for repository siting. Stress levels need to be determined and evaluated.

The seismicity of the Salina basin is low. Few earthquakes have been recorded, and all were of low to moderate intensity. However, local effects of seismic events associated with the Clarendon-Linden fault zone may affect the western part of the expanded New York study area.

Very little is known about jointing in the study area. There is no known igneous activity in the Ohio study area.

B.4.5 Resource Factors--Salina Basin

For preliminary evaluations, salt under large surface water bodies such as the Great Lakes and the Finger Lakes in New York have been removed from further consideration. Water resources have been identified for the purpose of gathering information but are not considered relevant to the preliminary screening of the region since they consisted of local water supplies. Water-resource conflicts would become more important in the evaluation of specific locations.

In regard to mineral resources, the regional study identified a number of resources that may conflict with repository siting. Information on the historic and current exploitation of mineral resources within and near the salt basins provides a basis for avoiding conflicting usage by assessing the location and extent of artificial breaching of the salt beds.

New York has abundant surface and near-surface mineral resources. The exploitation of these shallow resources has no apparent effect on the salt strata in the study area. Salt mining, brining operations, and the activities of the petroleum industry, however, have affected the salt beds.

The New York area contains many historic and present salt mining and brining operations. Important underground salt mines are located in Livingston and Tompkins Counties. Major artificial brine fields have been located in Wyoming, Onondaga, and Schuyler Counties. Other salt-producing operations were concentrated in Genesee, Wyoming, Livingston, and Ontario Counties; counties along the southern edge of the study area have had very few or no salt operations.

Relatively few petroleum wells in New York have penetrated the Salina salt beds. Most oil wells have been drilled only to Upper Devonian strata, from which all oil production in New York has been obtained. Much of the natural gas produced in the State has come from the Devonian Oriskany and Onondaga Formations, which lie above the salt beds. Gas production from formations deeper than the salt has been confined mostly to far western New York and to a band extending from Genesee and Wyoming Counties through Onondaga County. These areas have the highest density of wells penetrating Silurian strata. The part of the study area with the fewest wells penetrating Silurian strata extends from Allegany County through Broome County, south of the Finger Lakes. All production there has been obtained from formations lying above the Silurian salt.

Eastern Ohio contains abundant mineral resources. Most of the surface and near-surface mineral deposits are mined by open-pit methods.

Approximately 300 coal mines in the study area produce from the Pennsylvanian system. About 30% of these are underground mines, and none are more than 400 to 500 ft deep. It is unlikely that the open-pit and shallow-mining operations have affected the Salina salt beds. The only deep mine in eastern Ohio (except for two salt mines) is in Barberton, Summit County. This mine is 2,300 ft deep and extracts limestones from the Middle Devonian Delaware and Columbus Formations well above the Salina salts. The possible effects of this mining operation on the Salina salt beds in the area have not been evaluated.

Two deep eastern Ohio salt mines, one in Lake County and one in Cuyahoga County, extract salt from beds at depths of 1,760 and 1,920 ft, respectively. Four artificial brine operations, two in Summit County and one each in Lake and Wayne Counties, extract salt from various units of the Salina. Three brine operations in Cuyahoga County and one in Medina County have been conducted in the past, with wells ranging from 1,950 to 2,707 ft in depth. The effect of these artificial brine operations on the Salina salt beds needs evaluation. In the 19th century, there were four natural brine operations in the study area, one each in Columbiana, Tuscaraws, Morgan, and Guernsey Counties; well depths ranged from 450 to 900 ft.

The oil and gas industry has had a direct effect on the Salina salt beds in Ohio. The Clinton sand of Silurian age underlies the Salina Group and is the major oil and gas producer in the study area. It is estimated that about 51,000 wells penetrate the Devonian Oriskany and the Silurian Newburg (Salina) and Clinton pay zones in eastern and central Ohio. Most of these wells reach the Clinton. An even greater number of wells have been drilled into Pennsylvanian and Mississippian pay zones, but they are not likely to have affected the underlying Salina salt.

The mineral resources in the Lower Peninsula of Michigan consist of glacial deposits, several varieties of rock, natural brines, bedded salt, oil, and gas. The glacial deposits, limestone, dolostone, sandstone, and most of the gypsum, are mined by surface methods. Two shallow underground mines exploit gypsum at Grand Rapids, and coal was extracted from underground mines and by surface methods in the Saginaw Bay area until 1952. All of these mining operations are too shallow and too high in the stratigraphic column to have affected the bedded-salt deposits.

However, one large salt mine beneath Detroit works the Salina salt at a depth of 1,130 ft. The Silurian salt beds are solution-mined in St. Clair and Wayne Counties in southeastern Michigan and in Munistee and Maron Counties in western Michigan. Both Salina and Detroit River salt beds are solution-mined in Midland County. The salt beds in areas of solution mining are affected by openings created by the wells, by cavities developed by the extraction of salt, and possibly by subsidence above the cavities. Natural brines in the Detroit River Group are pumped to the surface in Lapeer, Gratiot, Midland, Mainetee, and Mason Counties. Brines in Mississippian and Devonian strata are also presently exploited in several of these counties and formerly in several other counties as well. Most current natural brine and solution-mining operations are in areas of historic production.

There are about 700 oil and gas fields in the Michigan basin. Producing formations range from Ordovician to Mississippian in age. The fields in the middle of the basin produce from Mississippian and Devonian strata. Fields in Silurian beds are situated around the northern and southern margins of the basin, and fields in Ordovician strata are in southern and southeastern Michigan. More than 30,000 wells of all types have been drilled by the petroleum industry. Salt beds in areas of production will be affected by wells that penetrate them, and many old wells may be difficult to locate. Most of the oil and gas production in Michigan is obtained from Silurian reefs that lie beneath the salt beds. Future exploration is likely to be intense along the Silurian reef trend that extends from Mason County to Presque Isle County in the northern part of the basin and in an area extending from St. Clair County to Allegan County in the southern part of the state. The northern and southern parts of the Michigan study area include large areas where historic and current oil and gas fields present no obstacle to the siting of a repository.

In the preliminary screening of New York and Ohio, it was assumed that surface or near-surface mineral deposits like sand, gravel, peat, limestone, and sandstone were sufficiently widespread not to have a major bearing on the definition of areas for further investigation. The locations of deep mines, however, were taken into account because of the resources they contain and because of their direct effect on the salt deposits. Oil and gas fields active now or in the past were also considered to constitute a significant resource conflict. Some oil and gas fields have also had a direct effect on the salt deposits because of penetration by wells.

In the initial screening, areas within 6 miles of past or present salt mines or brine fields were deemed inappropriate for repositories. Areas within 3 miles of known gas fields were also screened out. Individual oil or gas wells not located within a producing field were not screened out but must be evaluated further as the siting investigations focus on potential repository locations.

Although considerable mineral and petroleum resources are present in various regions throughout the basin, there remain many areas that could satisfy the siting criteria concerning minimal resource conflict.

B.5 Basalt Waste Isolation Project

B.5.1 Summary

The Basalt Waste Isolation Project (BWIP) is evaluating the Department's Hanford Site in the State of Washington to determine whether it contains a suitable location for a repository in basalt. The Hanford Site, 576 square miles in area, is in the center of the Pasco basin (Figure B-13). Geologic and hydrologic investigations have been under way since 1977 and represent a continuation of an effort conducted between 1968 and 1972. Most of the results of these investigations have been published (41, 42).

Current data and understanding indicate that basalt is a suitable medium for the disposal of radioactive waste. Investigations of the Hanford Site to date indicate that it has suitable geologic and structural characteristics. The location and the movement of water in the unconfined and upper basalt confined aquifers are well understood. Questions about the location and movement of the water in the interbeds and interflows of Wanapum and Grande Ronde Basalts are being addressed and should be resolved in the next 2 to 3 years. The tectonic conditions of the area have been investigated, and it appears that the area is sufficiently stable for siting a repository. Current information and understanding indicate no conflicts with natural resources.

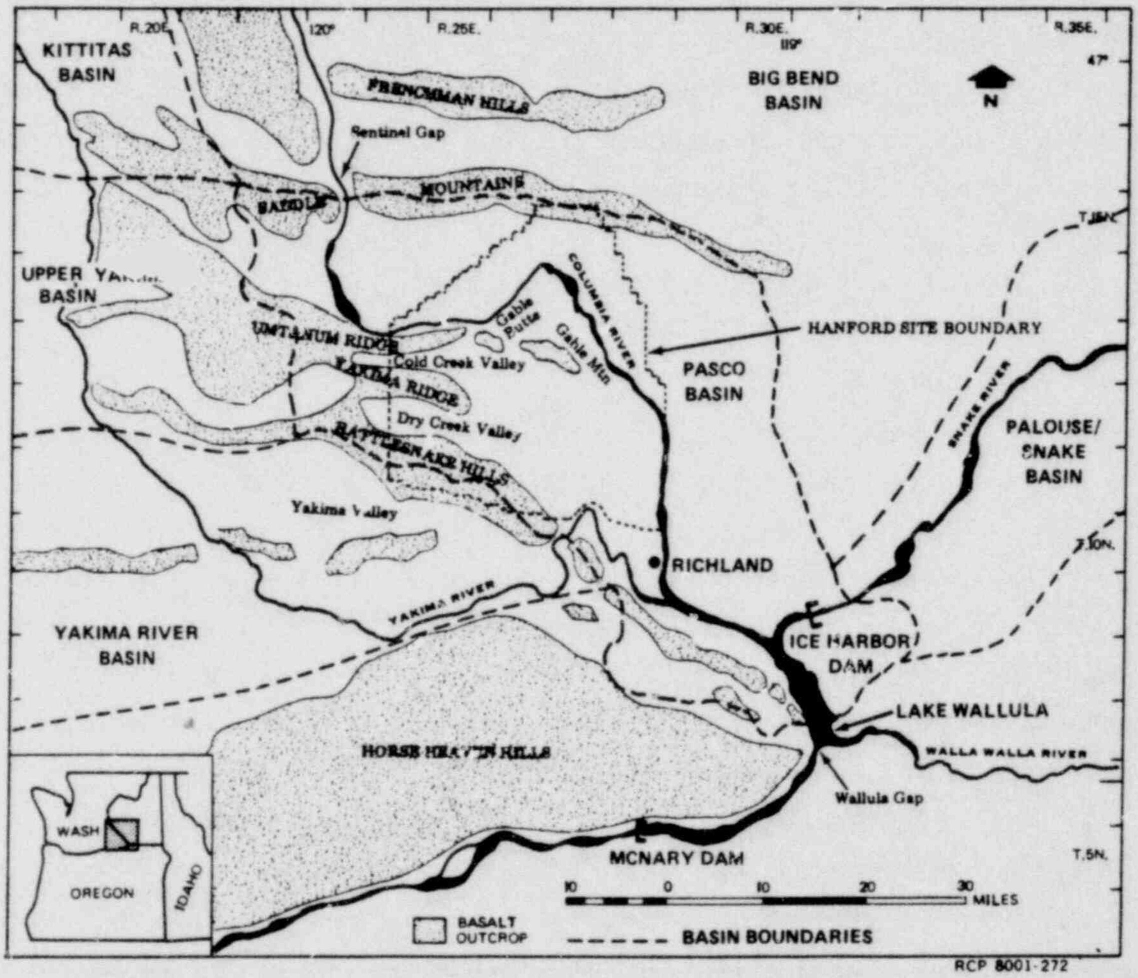


Figure B-13. Location of the Hanford Site in the Pasco Basin.

Source: (Reference 43) R.C. Edward, Geophysical Surveys in the Pasco Basin, RHO-BWI-79-100, Rockwell Hanford Operations, Richland, WA, 1979

B.5.2 Geologic Factors--Hanford Site

Yakima folds are the major geologic structures that have deformed the Columbia River basalt flows in the Pasco Basin since their emplacement and cooling (41). Individual Yakima folds trend east in the northwestern part of the basin and southeast in the southeastern part of the basin. Structural relief along the folds decreases to the east, but varies locally. The folds are locally faulted in some cases. Seismic reflection and aeromagnetic surveys are being used to trace small structures on the buried limbs of the fold and possible northwest-trending structures that cross-cut the folds (43).

Basalt stratigraphic relationships (thickness, sequence, and lateral extent) have been determined through the use of basinwide correlations combined with surface mapping and deep core drilling. Figure B-14 shows the stratigraphy of the Pasco Basin. Basalt unit identifications were made from chemical composition, paleomagnetism, and borehole geophysical logs. The Columbia River basalt in the Pasco Basin contains at least two, and possibly more, thick flows that are interpreted as being laterally continuous throughout the basin. These flows are tilted from the horizontal by less than 5 degrees across areas larger than 10 square miles beneath portions of the Hanford Site (41).

The candidate host rocks are mostly aphyric, with intersertal to intergranural textures. The chief constituents are plagioclase, clinopyroxene (augite, subcalcic augite, and pigeonite), and glass (the glass content varies internally within the flows). Details of the chemical and mineral composition have been documented (44, 45).

For a repository in basalt at 3,000 ft below the ground, the estimated maximum in situ stress (except for local concentrations) would be on the order of 4.0×10^3 psi. The average uniaxial compressive strength of the intact rock is 2.9×10^4 psi (45). The rock-mass properties of basalt --particularly the strength and, to a lesser degree, the thermal properties-- are affected by jointing. Fracture frequencies range from 6 to 10 fractures per meter, with rock-quality designations of 70% to 95% for the dense, central portions of deep basalt flows. The basalt flow that is considered to have the most desirable features for a repository is the Untanum flow. It averages

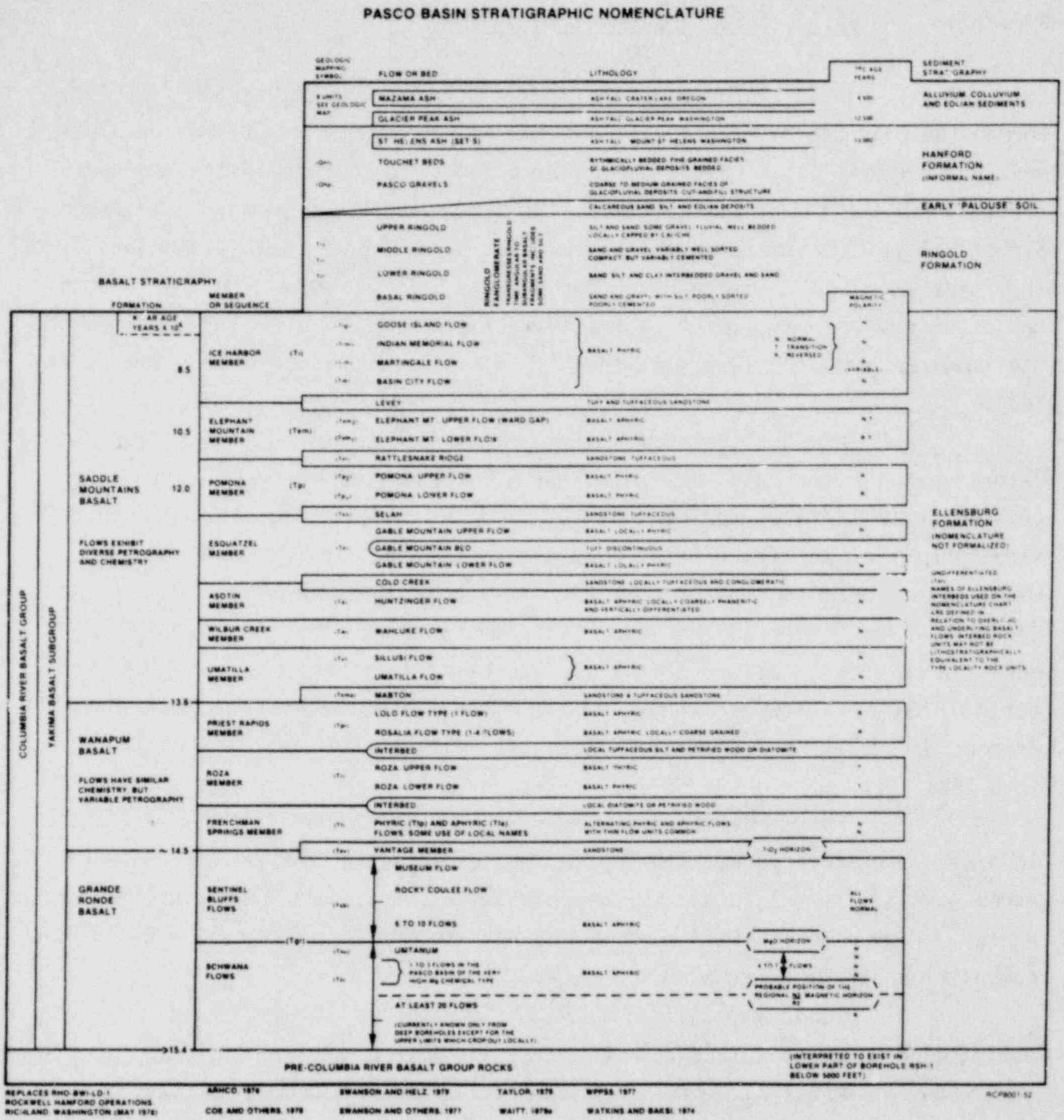


Figure B-14. Stratigraphic Relationships of the Columbia River Basalts in the Pasco Basin

Source: (Reference 41) C.W. Myers et al., Geologic Studies of the Columbia Plateau: A Status Report, RHO-BWI-ST-4, Rockwell Hanford Operations, Richland, WA, 1979 about 200 ft in total thickness across the central basin and has a dense interior that ranges from 150 to 175 ft in thickness.

A geothermal gradient of about 2.2°F per 100 ft of depth is estimated, based on a mean surface temperature of 53°F and a temperature of 120°F at a depth of 3,000 ft (46).

B.5.3 Hydrologic Factors--Hanford Site

The hydrologic evaluation of the Hanford Site is based on measurements made in 5 deep boreholes, 19 intermediate boreholes, and 300 shallow wells. In the Pasco Basin, the Columbia River Basalt Group is composed of three major basalt units, the Saddle Mountains Basalt Flows, the Wanapum Basalt Flows, and the Grande Ronde Basalt Flows (47). The basalt flows are covered with approximately 250 ft of alluvium, and the water table is approximately 150 ft below the surface (42, 48). Confined aquifers exist in the more permeable interflows and interbeds of these basalts. The major water-producing formations in the uppermost Saddle Mountain Basalts are the Mabton (sandstone and tuffaceous sandstone), Cold Creek (sandstone, locally tuffaceous and conglomerate), Selah (sandstone and tuffaceous), and Rattlesnake Ridge (sandstone and tuffaceous) interbeds. The relative positions of these interbeds in the basalt flows can be seen in Figure B-14. Within the intermediate-level Wamapum Basalts, the Vantage Sandstone is the only formation that produces significant quantities of water. In the deep Grande Ronde Basalts, only one basalt flow has been identified as producing significant quantities of water. This formation is several hundred feet below the Untanum flow but has not been named at this time.

The Saddle Mountains Basalt aquifers are recharged by the direct infiltration of precipitation and stream runoff in the Rattlesnake Hills, Yakima Ridge, and the Saddle Mountains (49). There is also evidence that recharge occurs where the unconfined aquifer and the Rattlesnake Ridge confined aquifer system of the Saddle Mountains Basalt (the uppermost confined aquifer) are in hydraulic communication beneath portions of the Hanford Site (50).

The principal recharge to the Wanapum and Grande Ronde Basalts in the eastern Pasco basin is from ground water inflow from the Palouse/Snake and Big Bend basins shown in Figure B-13. The Saddle Mountains Basalt forms the upper basalt formation in the eastern Pasco basin area and receives from the Columbia basin irrigation project most of the irrigation recharge that is not removed as agricultural return flow or unconfined aquifer recharge.

The major discharge area for the ground water of the Saddle Mountains Basalt is considered to be the Columbia River between Richland and Wallula Gap in the Horse Heaven Hills (see Figure B-13). This is more than 20 miles south of any candidate site now being considered for a repository beneath the Hanford Site. Here, the Saddle Mountains Basalt is in direct contact with the Columbia River, which has a surface water elevation of 340 ft above the mean sea level. Wanapum and Grande Ronde Basalts are known to crop out in the Snake River Gorge upstream from Ice Harbor Dam and the Yakima River Gorge north of Yakima, Washington, and in the Columbia River Gorge north of the Hanford Site. They also crop out along the fringes of the Columbia Plateau. The Snake River is a potential discharge area for the Wanapum and Grande Ronde Basalts east and southeast of the Pasco basin. Ground water discharge from these basalts is also to the Columbia River, probably in Lake Wallula or farther south in the Columbia River Gorge west of Lake Wallula, where the Wanapum and Grande Ronde basalts are in hydraulic communication with the river.

The hydraulic head gradient within the Mabton interbed of the Saddle Moutain Basalt is approximately 5 ft/mile. Ground water flows generally in the east-southeast direction within this formation beneath the Hanford Site; this trend is also supported by hydrochemical and isotopic data.

The hydraulic conductivity of the principal sedimentary interbeds (Rattlesnake Ridge, Selah, Cold Creek, and Mabton) within the Saddle Mountains Basalt is between 10^2 and 10^{-1} ft/day. Interflows of vesicular, brecciated basalt have hydraulic conductivities of 10^4 to 10^{-5} ft/day. Conductivities for the sections of columnar basalt (colonnade and entablature portions) are between 10^{-3} and 10^{-8} ft/day (42). The dense sections of the thick basalts such as the Untanum units within the Grande Ronde Basalt Flow have a hydraulic conductivity of about 10^{-8} ft/day. Formations with hydraulic conductivity from 10^5 to 10 ft/day are considered to be good or productive aquifers. Conductivities that range from 1 to 10^{-4} ft/day indicate that an aquifer will be poor or nonproductive. Rocks with a conductivity below 10^{-4} ft/day, while they do permit flow, are generally considered to be impervious.

Storage coefficients for the interbeds and the interflows are between 10^{-3} and 10^{-5}. These storage coefficients indicate that the interbeds and interflows are confined aquifers. If they were unconfined, the storage coefficient would be closer to 0.01 to 0.03.

The major inorganic constituents (see Table B-2) in ground water for the Columbia River Basalt Group in the Pasco Basin indicate that ground water within the Saddle Mountains and upper Wanapum Basalts is hydrochemically similar and is of a sodium bicarbonate chemical type. Ground water samples available from the Grande Ronde Basalt are of a sodium chloride, bicarbonate chemical type and thus are chemically distinct from the water in the other two basalt formations. In addition, comparisons of the average chemical compositions of the water show that the Grande Ronde Basalt has higher dissolved-solids concentrations (particularly chloride, sulfate, fluoride, sodium, and silica) than ground water in the Saddle Mountain or Wanapum Basalts. This is interpreted as strong evidence that the Saddle Mountain and Wanapum Basalts are not hydrologically interconnected with the Ground Ronde Basalts.

The age of the ground water is being investigated by carbon-14 age dating techniques. Water samples have been obtained from the Mabton interbed at eight different locations. The results indicate that the water is 12,000 to 24,000 years old (42). The youngest ground waters occur near recharge areas of the Rattlesnake Hills to Untanum Ridge. They become increasingly older with increasing distance from the recharge area, in the east and southeast direction. These data are consistent with the current concept of ground water flow developed from hydraulic head and hydrochemical data. Field testing is currently under way to date ground water in the interbeds and interflows of the deeper Wanapum and Grande Ronde Basalts.

B.5.4 Tectonic Factors--Hanford Site

Intrusive plutonic igneous rocks have not been detected within about 75 miles of the central Hanford Site (41). Intrusive basalt dikes are

Table B-2. Ground Water Chemistry of Basalt

Average Composition and Range in Concentration of
Major Chemical Constituents Within Ground Water for
Formation of Columbia River Basalt Group
(concentrations in mg/l)

Constituent	Lower Saddle Mountains Basalt	Upper Wanapum Basalt	Grande Ronde Basalt
ANIONS			
HCO_3	217 (169-267)	177 ([illegible]1-216)	75 (66-88)
$CO_3^{=}$	0	0	50 (101-127)
Cl^-	20 (4.3-63)	6.6 (3.8-15)	131 (98-148)
SO_4	4.0 (.3-18)	11 (.2-32)	72 (13-108)
NO_3	.5 (.5)	.5 (.1-2.7)	N.D.[a]
F^-	2.2 (.1-8.0)	.7 (.2-2.0)	29 (22-37)
CATIONS			
Na^+	83 (36-122)	34 (17-80)	225 (182-250)
K^+	11 (7.7-14)	11 (5.9-19)	2.5 (1.9-3.3)
Ca^{++}	4.7 (.5-22)	17 (1.6-24)	1.1 (.8-1.3)
Mg^{++}	1.8 (1.-12)	8.8 (.2-15)	.7 (.0-2.0)
SiO_2	69 (56-91)	57 (41-73)	117 (115-121)
Total dissolved solids (sum)	413 (344-505)	324 (283-435)	705 (584-826)

[a]N.D. = not detected.

Source: (Reference 130) R.E. Gephart et al., Hydrologic Studies Within the Columbia Plateau, Washington - An Integration of Current Knowledge, RHO-BWI-ST-5, Rockwell Hanford Operations, Richland, WA, October 1979 present about 35 miles east of the central Hanford Site, and the possibility cannot be dismissed that basalt dikes buried by subsequent basalt flows are in the central basin. This is under study. The most recent volcanic flow activity in the vicinity of the Hanford Site was approximately 5 million years ago. The closest approach of any portion of this flow to the central Hanford Site was approximately 15 miles to the southeast of the boundary of the site.

Geologic evidence from the regional and Pasco basin studies suggests that some tectonic deformation began as early as 14 to 15 million years ago. This deformation was localized along Yakima Fold trends and northwest-trending structures. The rate of tectonic deformation (i.e., the production of structural relief) increased locally along these same trends, but it was generally subdued 13.6 to 14.5 million years ago and was highest in the Pasco basin between 10.5 and 5 million years ago.

The occurrence of earthquakes in the Columbia Plateau indicates that the area continues to the experience relief of stresses, presumably tectonic. Focal mechanism solutions suggest that the maximum principal stress is nearly horizontal and oriented approximately north to south. Ten years of earthquake monitoring by a seismic network surrounding the Hanford Site indicates that earthquake activity in the area of the Pasco Basin appears to consist of (i) shallow, low-magnitude events in small volumes of basalt and (ii) deeper events below basalt that are of low magnitude and generally diffuse (51). Typical swarms contain events ranging in Richter magnitude from 0 to 4.0, but the magnitude of most of the events is less than 2.0.

The tectonic stability of the Pasco basin has been investigated, and the rate and nature of deformation along existing structures within the basalt have been studied. Five surveys of a trilateration array between 1972 and 1979 revealed no statistically significant deformation.

B.5.5 Resource Factors--Hanford Site

Columbia River basalts are notable for their relative lack of valuable mineral resources. No valuable mineral or hydrocarbon resources are known to have been produced in appreciable amounts from the Pasco Basin, with the exception of small quantities of natural gas produced in the 1920's and early 1930's from wells penetrating interbed sediments between Columbia River basalt flows, approximately 15 miles south of the central Hanford Site. Leasing by petroleum companies in eastern Washington adjacent to the west and northwest boundaries of the Hanford Site indicates interest in further investigating the potential for hydrocarbon resources in sedimentary rocks beneath the basalt in the Pasco Basin, at depths of or below 10,000 ft.

Ground water resources in the vicinity of the site are being evaluated. Current information suggests that these ground water resources or their potential for future development would not affect the integrity of the repository, especially since more readily available surface and ground waters are available elsewhere in the area.

Limited data suggest that heat flow in the Pasco basin ranges from 1 to 1.5 heat-flow units compared with a crustal average of about 1.0 to 1.5 units. The geochemistry and temperature of artesian spring water immediately to the west of the Hanford Site and geologic studies suggest that the geothermal resource potential of the proposed repository area is extremely low.

B.6 Nevada Nuclear Waste Storage Investigations

B.6.1 Summary

The Nevada Nuclear Waste Storage Investigations (NNWSI) are evaluating the suitability of the Department's Nevada Test Site (NTS) in Nye County, Nevada (Figure B-15). Because waste isolation activities must not interfere with the prime mission--nuclear weapons testing--NTS exploration for a suitable repository site on the NTS is currently limited to the southwest portion (Figure B-16). The area compatible with weapons testing consists of approximately 300 square miles of desert basins and mountain ranges.

Early in the project, locations outside the southwest quadrant were considered. One such location, the Syncline Ridge, was eliminated from further consideration because of its structural complexity (52). In addition to exploration for repository sites, field tests are being performed at the Climax Stock and G-Tunnel, to evaluate the suitability of granite and welded tuff, respectively, as host media. A summary of the investigations conducted to date is available in a recent report (53).

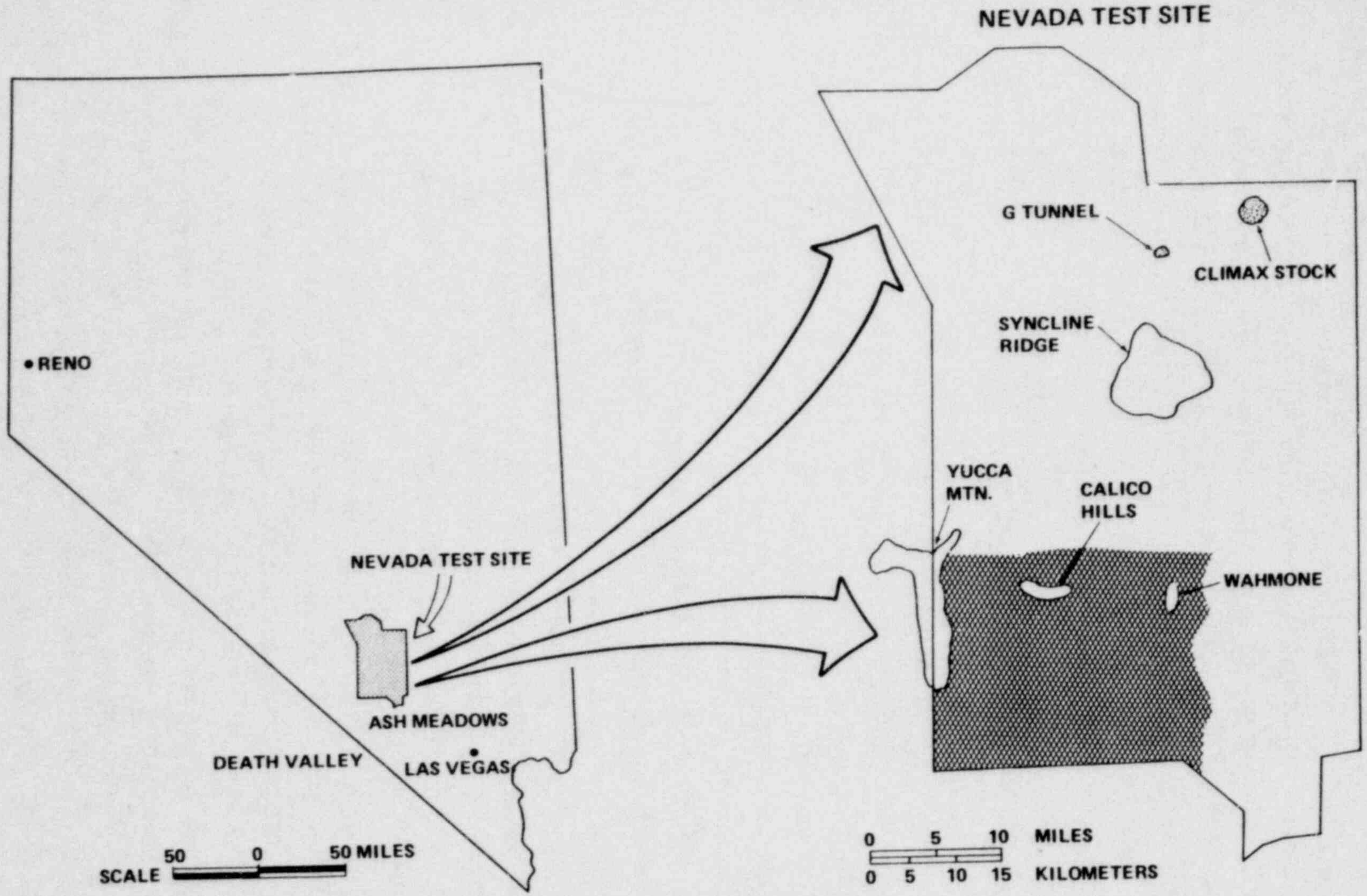


Figure B-15

Location of the Nevada Test Site
In Nevada

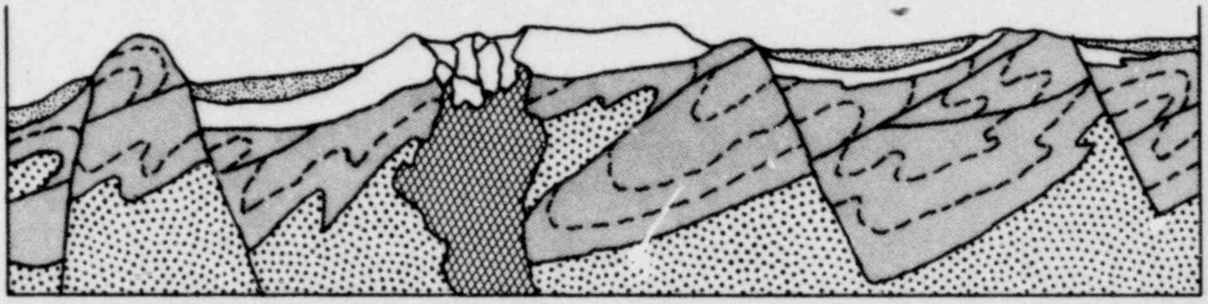
Current Exploration on Areas (Shaded)
and Supporting Test Facility Areas
(Stippled) at the Nevada Test Site

Current data and understanding indicate that the Nevada Test Site and tuffaceous rocks have potential for isolating radioactive waste in a mined geologic repository. The identification of a site with suitable geologic and hydrologic characteristics is the primary focus of exploration. The likelihood and the consequences of potential tectonic events are being analyzed to determine whether the tectonic setting would preclude waste disposal at the site. Preliminary data suggest that the risks from tectonic phenomena are acceptable. Metallic and energy-resource conflicts are minimal; both water and land-use resources need careful evaluation.

B.6.2 Geologic Factors--Nevada Test Site

The geology of the Nevada Test Site is complex, a characteristic shared by all of the Basin and Range Province in which the NTS is located. The geologic strata present at the NTS consist of more than 30,000 ft of miogeosynclinal pre-Cambrian and Paleozoic quartzites, shales, and carbonates (54). This sequence of sedimentary rocks was folded, thrust faulted, and intruded by granites during compressional mountain-building episodes in Mesozoic time. During the mid-Cenozoic, a few thousand feet of silicic and minor basaltic volcanic deposits were deposited on the eroded remnants of the Mesozoic Mountains and subsequently displaced by normal faults (54, 55). The closed basins associated with the normal faulting have accumulated alluvium deposits that exceed 2,000 ft in thickness in the deepest parts of the basins (56). The Tertiary alluvial basin deposits are interbedded with minor basalt flows, dikes, and sills. Figure B-16 shows the general stratigraphic and structural conditions of the NTS region.

Of the rock types that occur at the Nevada Test Site, argillite, granite, alluvium, and tuff have been considered for suitability as host rocks. Alluvium was deferred from consideration as a candidate host for high-level waste because its thermal conductivity (assumed for the study to range from 0.2 to 1.2 W/m-K) would allow unacceptable near-canister temperatures for 10-year-old high-heat-generating wastes (56). Geophysical data collected during 1978 and 1979 indicated the presence of structural discontinuities passing through the Calico Hills (argillite-granite) and Wahmonie (granite) study areas. Magnetic and gravity data suggested a possible granitic intru-



LEGEND

Tertiary - Quaternary Alluvium

Tertiary Volcanic Deposits

Tertiary Intrusions (?)

Precambrian and Paleozoic Quartzites, Carbonates and Shales

Precambrian Shield (?)

Figure B-16. Idealized Stratigraphic and Structural Relations in the Nevada Test Site Region sion at shallow depth below the argillites at Calico Hills, but a 2,550-ft drill hole failed to penetrate the inferred granitic mass (57). Therefore, current exploration efforts in the southwestern part of the Nevada Test Site are directed to locations containing the remaining candidate host rock, volcanic tuff. At present, only one location, Yucca Mountain (Figure B-15), is being explored.

Yucca Mountain is underlain by approximately 2,000 m of interbedded welded to nonwelded tuffs. The thermal, mechanical, and chemical properties of welded tuff are generally considered favorable for a geologic repository (58). The thermal conductivity of saturated welded tuff is about 2.5 W/m-K, and its mechanical strength approaches that of granite (59). However, welded tuff may contain up to 10% water by weight. If this water is removed from the rock, thermal conductivity decreases to approximately 1.7 W/m-K. The effects of this water on the thermomechanical response of tuff have to be assessed and are being investigated by an at-depth heating test at G-tunnel (60) and by laboratory studies and computer modeling. The sorption capacity of tuff is generally very high and quite variable. Depending on the mineralogy, sorption values for various types of tuff range from 55 to 13,000 ml/g for strontium and from 50 to 6,000 ml/g for cesium.

An ideal geologic setting for a repository in tuff is a thermally conductive, mechanically strong, welded tuff enveloped by a low-permeability, highly sorptive, nonwelded zeolitized tuff (Figure B-17). Field mapping, core drilling, and geophysical surveying are in progress to assess the extent to which these conditions exist at Yucca Mountain. A 6,000-ft core and hydrologic test hole is being drilled into the study area; the results will be correlated with data from a 2,500-ft hole drilled earlier (61). The water-bearing properties of inferred fracture zones in the Yucca Mountain area will be evaluated by hydrologic testing and geophysical surveys.

B.6.3 Hydrologic Factors--Nevada Test Site

The Nevada Test Site lies in a portion of the Great Basin characterized by large closed ground water and surface water basins. Ground water flows southwest and discharges at Ash Meadows, Nevada, and Death Valley, California (Figure B-15), about 40 and 60 km from the current study area,

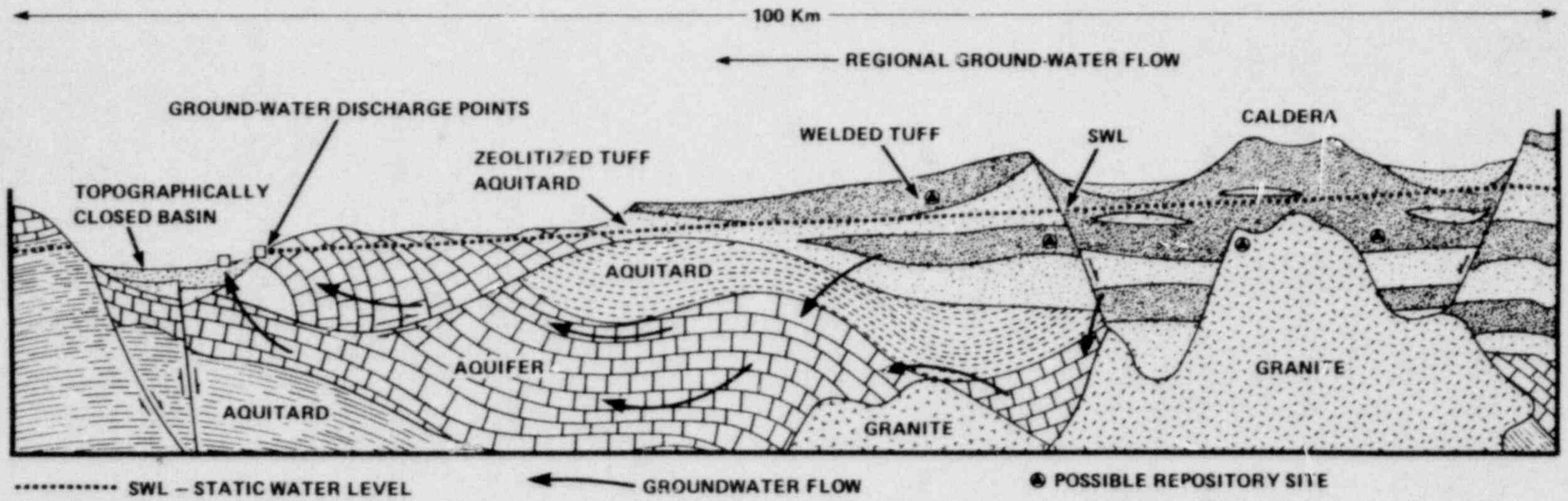


Figure B-17. Idealized Settings for a Repository in Tuff respectively (62). The water table is between 400 and 600 m deep in the study area (61, 62). The land surface slopes southward and intersects the potentiometric surface at Ash Meadows. Producing water wells at Lathrop Wells and the Amargosa Farms are downgradient from Yucca Mountain.

Few reliable estimates of ground water flow velocity are available for the NTS region. Estimates based on radiocarbon ages of water from wells at the NTS and from wells in the Amargosa Farms area, approximately 32 km downgradient, indicate an average velocity of about 9 m/yr between these wells (53). This rate is probably higher than average ground water velocities in the NTS because much of the flow to the wells is through local zones of fractured tuffaceous rocks and alluvium whose average hydraulic conductivity is higher than for other areas at the site with the exception of some areas of carbonates. Local flow rates in the carbonate rocks could be much higher (62). Additional data are needed before ground water velocities at the Nevada Test Site can be adequately characterized. Before the NNWSI project, the potential for off-site migration of radionuclides from weapons-testing activities had been evaluated (63). The sorption capacities of tuffaceous rocks and alluvium along flow paths from the study area are very high in comparison with most rock types (58).

A two-dimensional, finite-difference model of the regional flow system encompassing the Nevada Test Site is currently being tested by sensitivity analyses to determine the relative importance of the model's hydrologic parameters to velocity and residence-time calculations (64). The hydrologic model will be modified to include sorption mechanisms and capacities as well as radioactive decay chains for the radionuclides of interest.

Mapping of ancient spring deposits and analysis of clay minerals in alluvium indicate that ground water levels at discharge areas during pluvial episodes were never more than 200 ft higher than at present (65).

B.6.4 Tectonic Factors--Nevada Test Site

The Nevada Test Site is in an area of the Great Basin that has relatively low seismic activity for the Basin and Range Province (seismic risk zone 2) (66-68) and a low potential for volcanic eruptions (69, 70). Based on the historic seismic record within 400 km of the NTS, it is estimated that an acceleration of 0.7g has a return period of 15,000 years, and 0.5g has a return period of about 2,500 years (67). A net of about 45 seismometers has been deployed in a radius of approximately 150 km about the NTS to provide data for refining these estimates. In addition, fault scarps that offset alluvial deposits are being mapped and analyzed to study the history of Pleistocene and Holocene faulting and its relation to current and expected seismic activity.

Methods for estimating the recurrence potential of basaltic volcanism yield preliminary annual probabilities for a 10 km^2 area of 10^{-8} to 10^{-9} per square kilometer (70). The calculations assume that volcanism is a temporally and spatially random process within the NTS region. This assumption is conservative. Current work, including field and geophysical studies, are concerned with identifying structural features that affect the distribution of past volcanic activity. These structural controls will be factored into refined volcanic probability estimates. The youngest silicic volcanism in the vicinity of the Nevada Test Site is about 6 to 8 million years old, and it is unlikely that silicic volcanism would pose a hazard to a repository sited in the NTS region (71-74).

The relationship between Quaternary strata of the basin fill deposits and erosional and depositional surfaces indicates that only very slight regional and local uplift and subsidence have occurred during the past few million years (72-74). Therefore, it is assumed that the current regime of tectonic stability is very likely to persist over the next tens of thousands of years in the NTS area. The likelihood for isolated individual tectonic events is being evaluated. An assessment of the potential consequences of tectonic events is also under way.

B.6.5 Resource Factors--Nevada Test Site

Limited mining for gold, silver, mercury, and tungsten was conducted in the area of the Nevada Test Site long before the land was withdrawn for nuclear weapons testing. The mineral deposits are generally associated with intrusive granitic masses. All potential repository locations on or near the NTS will be evaluated for mineral resources.

No hydrocarbon deposits of reasonably foreseeable commercial grades occur at the Nevada Test Site. However, because the NTS is in a desert, water resources are scarce and must be considered.

Geothermal resources are not considered attractive at the Nevada Test Site. A general heat flow of about 1.5 to 2.0 heat flow units exists in the region.

B.7 Expanded National Waste Terminal Storage Program

The Department's site exploration program is being expanded to consider a wider variety of rock types in diverse geologic environments. These broadened activities were originally recommended by the Interagency Review Group and were included in the President's statement of 12 February 1980.

Three approaches to site exploration and characterization are currently available for use to implement the expanded program at the national screening phase. The approaches are (i) geologic--host rock; (ii) current land use; and (iii) geohydrologic environment. The 3 site exploration and characterization approaches are described in Section III.C.1.

The geologic, or host-rock, approach has been applied in separate literature surveys of granitic intrusive rocks in the conterminous United States and in the southern Piedmont to determine their distribution and potential suitability as repository host rocks (75, 76). The factors evaluated included physical, chemical, hydrologic, tectonic, seismic, and mineralogic properties. The information collected in these studies indicates that numerous granitic bodies are suitable for further evaluation.

Three evaluations have been completed in the following subregions of the southeastern United States:

1. The Piedmont Province, consisting of pre-Cambrian and Paleozoic metamorphic and igneous rocks (76).
2. Triassic basins, long narrow troughs in the Piedmont of Triassic sedimentary and extrusive rocks.
3. The Coastal Plain, a wedge of unconsolidated and semiconsolidated sands, clays, and limestones overlying rocks of the Piedmont type (67).

A literature study of argillaceous rocks in the United States is currently under way. Laterally persistent argillaceous rock units at least 75 m thick and 300 to 1000 m deep are being evaluated. Characteristics being considered include physical properties, mineral composition, geochemistry, geologic structure, seismic potential and tectonic history, development of mineral resources, regional ground water hydrology, and the extent of drilling and mining. This study is expected to be completed in the third quarter of fiscal year 1980.

Three broad classes of argillaceous rocks are being evaluated by laboratory testing:

1. Argillaceous rocks with no carbonaceous material and with little or no hydrous expandable clay.

2. Argillaceous rocks with carbonaceous material.

3. Argillaceous rocks with smectite as a major clay mineral constituent.

Each class will be sampled and evaluated to determine the major mineralogic constituents. Samples will be obtained by drilling. Each sample will be characterized by (i) complete chemical and mineralogical analyses; (ii) response or reaction to large doses of gamma radiation; (iii) measurement of the thermal properties; (iv) measurement of absorptive properties; (v) measurement of the mechanical properties; and (vi) determination of the products of the interaction between the samples and simulated radioactive waste. This project will continue through fiscal year 1981.

Screening based upon consideration of geohydrologic environments is being initiated to evaluate simultaneously the literature regarding geohydrologic and environmental requirements. This study will entail a nationwide literature search and result in a computerized data base suitable for use in parametric analyses. Concurrently, the U.S. Geological Survey is planning evaluations of various geohydrologic provinces of the United States.

References for Appendix B

(1) A. Holmes and L. Holmes, Principles of Physical Geology, Third Edition, Halsted Press, New York, NY, 1978

(2) R.E. Anderson, D.H. Eargle, and R.O. Davis, Geologic and Hydrologic Summary of Salt Domes in Gulf Coast Region of Texas, Louisiana, Mississippi, and Alabama, Open-file Report USGS-4339-2, U.S. Geological Survey, 1973

(3) C.W. Kreitler, E.G. Wermund, E.H. Guevara, A. Giles, and G. Fogg, Preliminary Evaluation of Salt Domes in East Texas, NWTS Program, U.S. Department of Energy, 1978

(4) J.D. Martinez, R.L. Thomas, D.H.Kupfer, C.G. Smith, Jr., C.R. Kolb, E.J. Newchurch, R.E. Wilcox, T.A. Manning, Jr., M. Romberg, A.J. Lesis, and R.E. Rovik, An Investigation of the Utility of Gulf Coast Salt Domes for the Storage or Disposal of Radioactive Wastes, Institute for Environmental Studies, Louisiana State University, Baton Rouge, LA, 1976

(5) P.P. Mellen, Final Report, Preliminary Investigation of Mississippi Salt Domes, Report submitted to OWI, UCC Nuclear Division, Y 12-36, Y16508V, Oak Ridge, TN, 1976

(6) Netherland, Sewell and Associates, Geologic Study of the Interior Salt Domes of Northeast Texas Salt Dome Basin to Investigate their Suitability for Possible Storage of Radioactive Waste Material as of May, 1976, p. 55, Report to OWI, UCC Nuclear Division, Oak Ridge, TN, 1976

(7) J.O. Ledbetter, W.R. Kaiser, and E.A. Ripperger, Radioactive Waste Management by Burial in Salt Domes, AEC Contract AT 40-1-4639, Engr. Mechanics Research Lab., University of Texas, Austin, TX

(8) Office of Nuclear Waste Isolation Staff, NWTS Criteria for the Geologic Disposal of Nuclear Wastes: Site Qualification Criteria, ONWI-33(2), Battelle Memorial Institute, Columbus, OH, December 1979

(9) G.D. Brunton, R.B. Laughon, and W.D. McClain, Screening Specifications for Gulf Coast Salt Domes, Y/OWI/TM-4B, Office of Waste Isolation, Oak Ridge, TN, 1978

(10) NUS Corporation, Non-geologic Criteria for Radioactive Waste Repositories, Y/OWI/SUB77/16504/10(Draft), Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

(11) P.F. Patchik, The Suitability of Palestine Dome, Anderson County, Texas, for a High-Level Nuclear Waste Repository, ONWI-74, Battelle Memorial Institute, Columbus, OH, 1980

(12) R. Kehle, Tectonic Framework and History, Gulf of Mexico Region, Kehle, Ralph, and Associates, Austin, TX, 1978

(13) J.N. Payne, Hydrologic Significance of the Lithofaces of the Sparta Sand in Arkansas, Louisiana, Mississippi, and Texas, Professional Paper 569-A, U.S. Geological Survey, 1968

(14) Law Engineering and Testing Co., Inc., Geologic Evaluation of Gulf Coast Salt Domes, ONWI/Sub/894-2231C, (Draft), Battelle Memorial Institute, Columbus, OH, 1978

(15) C.E. Glass and D.B. Slemmons, State-of-the-Art for Assessing Earthquake Hazards in the United States, Miscellaneous Paper S-73-1, U.S. Army Engineers Waterways Experiment Station, 1978

(16) E.W. Collins, D.K. Hobday, and G.W. Kyertler, Quaternary Faulting in East Texas, TBEG Geologic Circular 80-1, Texas Bureau of Economic Geology, University of Texas, Austin, TX, 1980

(17) G.E. Murray, Mineral Resource Potential for Selected Salt in the Gulf Interior Region, LETCO Report, Law Engineering Testing Company, Marietta, GA, 1980

(18) K.S. Johnson and S. Gonzales, Salt Deposits in the U.S. and Regional Characteristics Important for Storage of Radioactive Waste, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, 1978

(19) R.J. Hite and S.W. Lohman, Geologic Appraisal of Paradox Basin Salt Deposits for Waste Emplacement, Open File Report 4339-6, U.S. Geological Survey, 1973

(20) Woodward-Clyde Consultants, Paradox Basin Geologic Exploration Area Characterization Plan, (Draft in review), Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH

(21) Woodward-Clyde Consultants, Regional Geology of the Paradox Basin, ONWI-92 (Draft in review), Battelle Memorial Institute, Columbus, OH

(22) Office of Nuclear Waste Isolation, Technical Progress Report in the Quarter April 1-June 30, 1979, p. 319, ONWI-9(3), Battelle Memorial Institute, Columbus, OH, 1979

(23) H.W. Merrell and staff of the Utah Geological and Mineral Survey, "Mineral Resource Inventory of the Paradox Salt Basin, Utah and Colorado," Utah Geological and Mineral Survey Report of Investigation No. 143, 1979

(24) T.C. Gustavson et al., Locating Field Confirmation Study Areas for Isolation of Nuclear Waste in the Texas Panhandle, ONWI/79/E-511-00300-4, Bureau of Economic Geology, University of Texas at Austin, Austin, TX, March, 1980

(25) K.S. Johnson, Evaluation of Permian Salt Deposits in the Texas Panhandle and Western Oklahoma for Underground Storage of Radioactive Waste, Y/OWI/SUB-4494/1, Oak Ridge, TN, 1976

(26) S.P. Dutton et al., Geology and Geohydrology of the Palo Duro Basin, Texas Panhandle, pp. 87-95, the University of Texas at Austin, Bureau of Economic Geology, Geological Circular 79-1, Austin, TX, 1979

(27) U.S. Geological Survey, Water Resources Data for Texas, Part II: Water-Quality Records, U.S. Geological Survey Basic Data Report, 1969-1974

(28) R.Y. Anderson, Deep Dissolution of Salt, Northern Delaware Basin, New Mexico, University of New Mexico, prepared for Sandia National Laboratories, Albuquerque, NM, 1978

(29) D.W. Powers, S.J. Lambert, S.E. Shaffer, L.R. Hill, and W.D. Weart, Geologic Characterization Report for the Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico (2 volumes), SAND78-1596, Sandia National Laboratories, Albuquerque, NM, 1978

(30) U.S. Department of Energy, Draft Environmental Impact Statement - Waste Isolation Plant, DOE/EIS-0026-D, April 1979

(31) W.L. Hiss, Stratigraphy and Ground Water Hydrology of the Capitan Aquifer, Southeastern New Mexico and Western Texas, Ph.D. Thesis, University of Colorado, Boulder, CO, 1975

(32) G. O. Backman, Geologic Processes and Cenozoic History Related to Salt Distribution in Southern New Mexico, Open-File Report 74-194, U.S. Geological Survey, Denver, CO, 1974

(33) A.L. Brokow, Geology and Hydrology of the Carlsbad Potash Area, Eddy and Lea Counties, New Mexico, USGS OFR 4339-1, U.S. Geological Survey, 1972

(34) G.O. Bachman, Geologic Processes and Cenozoic History Related to Salt Distribution in Southeastern New Mexico, Open-File Report 74-194, U.S. Geological Survey, Denver, CO, 1974

(35) J.W. Mercer and B.R. Orr, Interim Data Report on the Geohydrology of the Proposed Waste Isolation Pilot Plant Site Southeast New Mexico, Water Resources Investigation Report 79-98, U.S. Geological Survey, July 1979

(36) D.W. Powers, S.J. Lambert, S.E. Shaffer, L.R. Hill, and W.D. Weart, Geologic Characterization Report for the Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico (2 volumes), SAND78-1596, Sandia National Laboratories, Albuquerque, NM, 1978

(37) A.R. Sanford, S.J. Johansen et al., A Study of Earthquakes in the Permian Basin, Texas-New Mexico, Geological Survey Draft Report, U.S. Geological Survey, 1978

(38) U.S. Department of Energy, Draft Environmental Impact Statement - Waste Isolation Pilot Plant, DOE/EIS-0026-D, April 1979

(39) Stone and Webster Engineering Corp., Regional Geology of the Salina Basin, Battelle/ONWI/Sub-E512-06001, (two volumes), Battelle Memorial Institute, Columbus, OH, 1978

(40) Stone and Webster Engineering Corp., New York: Ohio Technical Update and Summary, Battelle/ONWI/Sub-E512-00600/12, Battelle Memorial Institute, Columbus, OH

(41) C.W. Myers, S.M. Price, J.A. Caggiano, P.M. Cochran, W.J. Czimer, N.J. Davidson, R.C. Edwards, K.R. Fecht, G.E. Holmes, M.G. Jones, J.R. Kunk, R.D. Landon, R.K. Ledgerwood, J.T. Lillie, P.E. Long, T.H. Mitchell, E.H. Price, S.P. Reidel, and A.M. Tallman, Geologic Studies of the Columbia Plateau: A Status Report, RHO-BWI-ST-4, Rockwell Hanford Operations, Richland, WA, 1979

(42) R.E. Gephart, R.C. Arnett, R.G. Baca, L.S. Leonhart, and F.A. Spane, Hydrologic Studies Within the Columbia Plateau, Washington: An Integration of Current Knowledge, RHO-BWI-ST-5, Rockwell Hanford Operations, Richland, WA, 1979

(43) R.C. Edward, Geophysical Surveys in the Pasco Basin, RHO-BWI-79-100, Rockwell Hanford Operations, Richland, WA, 1979

(44) I.L. Wright, D.A. Swanson, R.T. Helz, and G.R. Byerly, Major Oxide, Trace Element, and Glass Chemistry of Columbia River Basalt Samples Collected Between 1971 and 1977, Open File Report 79-711, U.S. Geological Survey, 1979

(45) R.A. Deju, M.P. Baard, R.E. Gephart, and C.W. Meyers, "Structural Considerations in the Design of a Repository to Store Radioactive Waste in Basalt Formations," Proceedings of the International Tunnel Symposium, Tokyo, Japan

(46) Preliminary Feasibility of Radioactive Waste in Columbia River Basalts, ARH-ST-137, Atlantic Richfield Hanford Company, Richland, WA, November 1975

(47) D.A. Swanson et al., Reconnaissance Geologic Map of the Columbia Basalt Group, Pullman, and Walla Walla Quadrangles, Southeast Washington and Adjacent Idaho, Open File Map 77-100, U.S. Geological Survey, 1977

(48) Rockwell Hydrologic Science Unit, Hanford Site Water Table Map, Drawer #A2-38396, Rev. 8, Rockwell Hanford Operations, Richland, WA, December 1979

(49) R.E. Gephart, P.A. Eddy, and R.A. Deju, Geophysical Logging and Hydrologic Testing of Deep Basalt Flows in the Rattlesnake Hills Well Number One, RHO-BWI-ST-1, Rockwell Hanford Operations, Richland, WA, 1979

(50) R.K. Ledgerwood and R.A. Deju, Hydrogeology of the Uppermost Confined Aquifers Underlying the Hanford Reservation, ARH-SA-253, Atlantic Richfield Hanford Company, Richland, WA, 1976

(51) S.D. Malone, Annual Technical Report on Earthquake Monitoring of the Hanford Region, Eastern Washington, Geophysics Program, University of Washington, Seattle, WA, 1979

(52) J.B. Cotter, memorandum to D.L. Vieth, August 3, 1978, "Candidate Repository Site Suitability Determination, Control Block of Syncline Ridge Eleana Formation"

(53) NNWSI Project Officers, "A Status Report on the Evaluation of Nuclear Waste Repository Siting on the Nevada Test Site," p. 31, attached to November 21, 1979 letter from M.E. Gates to S. Myers, 1979

(54) E.B. Eckel, ed., "Nevada Test Site," Geol. Soc. of Amer. Memoir 110, p. 290, Boulder, CO, 1968

(55) F.M. Byers, et al., Volcanic Suites and Related Cauldrons of Timber Mountain - Oasis Valley Caldera Complex, Southern Nevada, Professional Paper 919, U.S. Geological Survey, 1976

(56) J.R. Smyth, B.M. Crowe, P.M. Halleck, and A.W. Reed, A Preliminary Evaluation of the Radioactive Waste Isolation Potential of the Alluvium-Filled Valleys of the Great Basin, p. 22, LA-7962-MS, Los Alamos National Scientific Laboratory, Los Alamos, NM, August 1979

(57) F. Maldonado, D.C. Muller, and J.N. Morrison, Preliminary Geological and Geophysical Data of Ue25a-3 Exploratory Hole, Nevada Test Site, Nevada, USGS-1543-6, U.S. Geological Survey, Denver, CO, September 1979

(58) L.D. Tyler, "Evanuation of Tuff as a Waste Isolation Medium," Proceedings of the Symposium on the State of Waste Disposal Technology, and the Social and Political Implications, Roy G. Post (ed.), February 26 - March 1, 1979, pp. 199-215, University of Arizona, College of Engineering, Tucson, Arizona

(59) G. Swan, The Mechanical Properties of the Stripa Granite, LBL-7074 and SAC-03, Lawrence Berkeley Laboratory, Berkeley, CA, 1978

(60) J.K. Johnstone, In-Situ Tuff Water-Migration/Heater Experiment: Experimental Plan, SAND79-1276, Sandia National Laboratories, Albuquerque, NM, 1980

(61) R.W. Spengler, D.C. Muller, and R.B. Livermore, Preliminary Report on the Geology and Geophysics of Drill Hole Ue25a-1, Yucca Mountain, Nevada Test Site, Open File Report 79-1244, U.S. Geological Survey, Denver, CO, 1979

(62) I.J. Winograd and W. Thordarson, Hydrologic and Hydrochemical Framework, South-Central Great Basin, Nevada-California, with Special Reference to the Nevada Test Site, Professional Paper 712-C, U.S. Geological Survey, 1975

(63) I.Y. Borg et al., Information Pertinent to the Migration of Radionuclides in Groundwater at the Nevada Test Site: Part 2, UCRL52078, Lawrence Livermore National Laboratory, Livermore, CA, 1976

(64) R.L. Iman, J.M. Davenport, R.K. Wadell, H.P. Stephens, and D.I. Leap, "Sensitivity Analysis of Parameters of a Regional Hydrologic Model Nevada Test Site and Vicinity, Nevada-California," Proceedings of the 1979 Department of Energy Statistical Symposium, October 1979, Gatlinburg, TN (in press)

(65) U.S. Department of Energy, Quarterly Report Nevada Nuclear Waste Storage Investigations, for quarter of July-Sept., 1979, NVO-196-12, December 1979

(66) S.T. Algermissen and D.M. Perkins, A Probabilistic Estimate of Maximum Ground Acceleration in the Contiguous United States, Open File Report 76-416, U.S. Geological Survey, 1976

(67) A.M. Rogers, D.M. Perkins, and F.A. McKeown, "A Preliminary Assessment of the Seismic Hazard of the Nevada Test Site Region," Bull. Seis. Soc. of Amer., Vol. 67, No. 6, pp. 1587-1606, December 1977

(68) A. Ryall, D.B. Slemmons, and Geoney, "Seismicity, Tectonism, and Surface Faulting in the Western United States During Historic Time," Bull. Seis. Soc. Amer., Vol. 56, pp. 1105-11[illegible], 1966

(69) D.R. Mullineaux, "Preliminary Overview Map of Volcanic Hazards in the 48 Conterminous United States," Scale 1:7,500,000 Miscellaneous Field Studies Map MF-786, U.S. Geological Survey, reprinted 1977

(70) B.M. Crowe and W.J. Carr, Pliocene and Quaternary Volcanic Rocks of the Southern Great Basin: Status of Volcanic Risk Assessment for the Nevada Nuclear Waste Storage Investigations, Open File Report, U.S. Geological Survey, Denver, CO, 1980

(71) B. M. Crowe and K.A. Sargent, Major-Element Geochemistry of the Silent Canyon-Black Mountain Peralkaline Volcanic Centers, Northwestern Nevada Test Site: Application to an Assessment of Renewed Volcanism, p. 25, Open File Report 79-926, U.S. Geological Survey, 1979

(72) R.C. Lincoln, attachments to memorandum to R.M. Nelson, "Tectonics Peer Review Meeting," October 15, 1979

(73) M.G. Best and K.W. Hamblin, "Origin of the Northern Basin and Range Province: Implications from the Geology of its Eastern Boundary," in Cenozoic Tectonics and Regional Geophysics of the Western Cordillera, Geol. Soc. of Amer. Memoir 152, R.B. Smith and G.P. Eaton (eds.), pp. 313-340, Boulder, CO, 1978

(74) R.L. Armstrong, E.B. Ekren, E.H. McKee, and D.D. Noble, "Space Time Relations of Cenozoic Silicic Volcanism in the Great Basin of the Western United States," Amer. Jour. Sci., Vol. 267, No. 4, pp. 478-490, April 1969

(75) Dames and Moore, Review of Potential Host Rocks for Radioactive Waste Disposal in the Southeastern United States Triassic Basin Subregions, Office of Waste Isolation, Oak Ridge, TN (undated)

(76) Ebasco Services, Inc., Review of Potential Host Rock for Radioactivity Waste Disposal in the Southeast United States, Southeastern Coastal Plain Subregion, Savannah River Laboratory, Aiken, SC, 1978

APPENDIX C
GLOSSARY OF KEY TERMS AND ACRONYMS

absorption:
see sorption.

accessible environment:
those portions of the environment directly in contact with or readily available for use by human beings. Includes the Earth's atmosphere, the land surface, surface waters, and the oceans. Also includes aquifers which have been designated as underground sources of drinking water (under 40 CFR 146) and are more than 1 mile from the original emplacement of the radioactive wastes.

actinides:
any radioactive element with an atomic number above 88.

activation:
the process of making a material radioactive by bombardment with neutrons, protons, or other nuclear particles.

activity:
a measure of the rate at which radioactive material is emitting radiation, usually given in terms of the number of nuclear disintegrations per unit of time. The standard unit of activity is the curie (Ci).

ACVSF:
air-cooled vault storage facility. An alternative for the extended dry storage of packaged spent fuel.

adsorption:
see sorption.

AFR:
away-from-reactor storage. Storage of spent fuel assemblies at a location not adjacent to the reactor in which they were used.

aging:
storage for the purpose of providing time for the decay of short-lived radionuclides.

ALARA:
as low as reasonably achievable. ALARA refers to limiting release and exposure and is used by the NRC (10 CFR 50.34) in the context of " . . . as low as reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations. . ."

alpha particle:
a positively charged particle, consisting of two neutrons and two protons, emitted by certain radioactive materials.

ANSI:
American National Standards Institute.

APS:
American Physical Society.

aquifer:
a layer of permeable rock or soil through which water flows.

aquitard:
a layer of permeable rock or soil that retards, but does not prevent, the flow of water to or from an aquifer.

argillaceous:
containing or pertaining to clay.

atomic number:
the number of protons within a nucleus.

atomic weight:
the mass of an atom relative to other atoms.

backfilling:
the placement of originally removed or new materials into excavated areas of a mine, including holes drilled for waste canisters, drifts, accessways and shafts.

background radiation:
radiation occurring naturally in the environment, including cosmic rays, the naturally radioactive elements of the Earth, and radiation from the human body itself.

barriers:
features of a waste disposal system which act to either contain or isolate radioactive waste.

beta particle:
a charged particle emitted by certain radioactive materials, which is physically identical to the electron.

biosphere:
the part of the Earth in which life can exist, including the lithosphere, hydrosphere, and atmosphere. See also: accessible environment and geosphere.

biota:
animal and plant life.

BLM: Bureau of Land Management, U.S. Department of the Interior.

breccia:
rock fragments cemented together in a fine-grained matrix.

burial grounds:
areas designated for near-surface disposal of low-level radioactive wastes.

burnup:
a measure of reactor fuel consumption, expressed either as the percentage of fuel atoms that have undergone fission, or the amount of energy produced per unit weight of fuel in the reactor.

BWIP:
Basalt Waste Isolation Project.

BWR:
boiling water reactor. A reactor system that uses boiling water as the coolant.

CA:
construction authorization. Authorization by the NRC to construct a nuclear facility.

calcine:
material heated to a temperature below melting point to bring about a more chemically stable form through oxidation and the loss of moisture.

canister array:
the geometric configuration of encapsulated spent fuel elements; usually associated with storage.

CFR:
U.S. Code of Federal Regulations.

chemical resynthesis:
the process whereby thermodynamic equilibrium between nuclear waste and its host rock is attempted in order to enhance waste form stability.

cladding:
a layer of metal or metal alloy surrounding the nuclear fuel, which provides a barrier between the fuel and the reactor coolant.

clastics:
rock or sediments composed principally of broken fragments that are derived from preexisting rocks or minerals.

confining unit:
a distinct hydrogeologic unit which neither transmits ground water readily nor yields significant quantities of water to wells or springs. See also: aquifer, aquitard.

constitutive law:
any law that defines natural relationships such as the dependence of strain on stress for solid materials or pressure-volume relationships for gases.

containment:
confining the radioactive wastes within prescribed boundaries, e.g., within a waste package.

containment integrity:
the ability of containment to meet applicable performance objectives.

control rod:
a device, insertable into a reactor core, which contains a material that readily absorbs neutrons, and which is used to control the power of a nuclear reactor.

control rod guide thimbles:
generally, the tube through which the control rods are inserted in a PWR fuel assembly.

convection cell:
an uneven movement of a stratum of the Earth's crust due to heat variations.

criticality:
a condition or state in which there occurs a self-sustaining nuclear chain reaction.

crud:
the normal buildup of corrosive products on fuel assembly surfaces.

curie (Ci):
a unit of measurement of activity. One curie equals that quantity of any nuclide which undergoes 3.7×10^{10} disintegrations per second.

DCSF:
dry caisson storage facility. An alternative for the extended storage of spent fuel.

decay:
the process whereby radioactive particles undergo a change from one isotope or state to another at a constant rate, releasing radioactive particles and/or energy in the process.

decommissioning:
removal from active service on a permanent basis.

decontamination:
chemical or mechanical removal of radioactive materials from nonradioactive materials in order that the latter would constitute a lesser radiation protection concern.

deep continental geologic formations:
geologic media beneath the continents, separated from biologic species and phenomena, and distinct from ice sheets and sea floor geologic media.

deionized water:
water which has undergone anion/cation exchange in order to remove ionic impurities.

Department:
U.S. Department of Energy (see also DOE).

diapirism:
process of piercing or rupturing of domed or uplifted overlying rocks by core material heated to the plastic state.

disposal:
the placement of radioactive waste in a manner which is considered permanent, with no intent to retrieve.

DOE:
U.S. Department of Energy.

dose commitment:
the total dose equivalent that results from an intake of radioactive materials during all the time from the intake to death of the organism. For humans, the dose is usually evaluated for a period of 50 years from the intake. Units are rem. See also: dose equivalent.

dose equivalent:
a basis for estimating consequential health risk, regardless of rate, quantity, source, or quality of the radiation exposure. Dose equivalent is expressed in units of rem.

DOT:
U.S. Department of Transportation.

drift:
a horizontal, or nearly horizontal, mined passageway.

EIA:
Energy Information Administration (of the U.S. Department of Energy).

Environmental Assessment (EA):

(a) A concise public document, for which a Federal agency is responsible, that serves to:
  (1) Briefly provide sufficient evidence and analysis for determining whether to prepare an environmental impact statement or a finding of no significant impact.
  (2) Aid an agency's compliance with the NEPA when an environmental impact statement is necessary.
  (3) Facilitate preparation of a statement when necessary.

(b) Shall include brief discussions of the need for the proposal, of alternatives as required by Sec. 102(2)(E) of NEPA, of the environmental impacts of the proposed action and alternatives, and a listing of agencies and persons consulted.

Environmental Evaluation (EE):
a brief document prepared by the Department to verify that no significant impact will result from proposed site characterization activities.

Environmental Impact Statement (EIS):
a detailed written statement as required by Sec. 102(2)(C) of the National Environmental Policy Act NEPA.

EPA:
U.S. Environmental Protection Agency.

epeirogeny:
slow uplift or subsidence over broad regions of the continental crust.

evaporites:
nonclastic sedimentary rocks composed principally of minerals produced from a saline solution that became concentrated by evaporation of the solvent. Rock salt and gypsum are examples of evaporites.

extensometer:
an instrument used in measuring strain.

fission products:
the nuclei formed by the fission of heavy elements, plus the nuclides formed by the subsequent radioactive decay of these fragments.

fuel:
fissionable material used as the source of nuclear power when placed in a critical arrangement in a nuclear reactor.

fuel assembly:
an array of fuel rods or plates in a fixed configuration.

fuel basket:
the holder or transfer mechanism for spent fuel in the water storage pool.

fuel cycle:
the series of steps involved in supplying fuel for nuclear power reactors, including mining, milling, and refining of uranium, the original fabrication of fuel elements, their use in a reactor, and possibly reprocessing and refabrication of the recovered material into new fuel elements.

fuel rod:
a tube into which fissionable material is sealed for use in a fuel assembly.

gamma flux:
the number of gamma rays per square centimeter per second.

gamma ray:
electromagnetic radiation, similar in nature to x-rays, emitted by the nuclei of some radioactive substances during decay.

GCR:
Geological Characterization Report.

GEIS:
a generic environmental impact statement.

geologic disposal:
placement of radioactive waste in deep stable geologic formations.

geometry control:
the method of preventing criticality in fissile material by controlling the configuration of the material, which includes limitation of the significant dimensions of vessels containing solutions of fissible material, and prevention by mechanical means of close approach to each other of discrete objects containing fissile material.

geosphere:
Any of the so-called spheres or layers of the Earth, including the atmosphere, lithosphere, and hydrosphere.

ground water:
water that exists or flows in or between geologic formations.

GWe:
gigawatts (electric). One gigawatt equals 1 billion watts of electricity.

half-life:
the time required for a substance to decay to half its original level of radioactivity.

health physics:
the discipline concerned with recognition, evaluation, and management of hazards from radiation.

HEPA filter:
high-efficiency particulate air filter. A filter designed to remove very small particulate matter from air with a high level of effectiveness.

HLW:
high-level radioactive waste; spent nuclear fuel; liquid wastes resulting from the operation of the first-cycle solvent extraction system and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel. Also, solids into which such wastes have been converted.

hot cell:
a heavily shielded enclosure in which radioactive materials can be handled by persons using remote manipulators and viewing through shielded windows or periscopes.

HTGR fuel:
high-temperature gas-cooled reactor fuel; used in graphite-moderated, helium-cooled reactors.

hydrosphere:
The waters of the Earth, as distinguished from the rocks (lithosphere), living things (biosphere), and air (atmosphere).

ICRP:
International Council on Radiation Protection.

IFSF:
independent fuel storage facility. Fuel storage facility located separately from a nuclear power plant or fuel reprocessing plant. See also: AFR.

igneous:
formed by volcanic action or intense heat (crystallized from an original melt).

immobilization:
treatment and/or emplacement of nuclear wastes so as to impede the movement of radioactive isotopes.

intrinsic permeability:
a measure of the relative ease with which a porous medium transmits a liquid under a potential gradient. It is a property of the medium alone and is independent of the nature of the fluid.

irradiated:
exposed to radiation (as from a nuclear reactor or particle accelerator). Spent fuel has been irradiated to the extent that it can no longer be used effectively in a nuclear reactor.

isolation:
segregating wastes from the accessible environment (biosphere) to the extent required to meet applicable radiological performance objectives.

K_d:
distribution coefficient. The concentration of material in solvent is divided by the concentration of material in solute.

K_{eff}:
effective neutron multiplication factor. A measure of the state of criticality of a reactor. For a nuclear reaction to be critical, the K_{eff} must be equal to 1.

laser interferometry:
a method using laser sources and reflecting mirrors for determining suitability of a particular portion of the Earth's crust for waste disposal.

lithosphere:
the solid portion of the Earth, as compared with the atmosphere and the hydrosphere; the crust of the Earth.

lithostatic pressure:
vertical pressure in the Earth's crust, equal to the pressure exerted by a column of the overlying rock or soil.

LLW:
low-level radioactive waste.

LWR:
light water reactor, which uses a coolant of ordinary water (H_2O) instead of heavy water (D_2O). An LWR may be either a BWR or a PWR.

metallography:
the study of the structure of metals, especially through use of the microscope.

microcurie/milliliter(μCi/ml):
unit of activity defined in terms of the number of disintegrations per second per unit quantity.

millirem/hour (mrem/hr):
the rate at which dose equivalent is received. A millirem is 1 one-thousandth of a rem. See also: dose equivalent.

mobility:
ability of a nuclide to migrate through a system.

multibarrier system:
a succession of barriers, operating independently or relatively independently, which act to contain and/or isolate nuclear waste.

MWd/mtu:
megawatt-day per metric ton of uranium. A unit used for expressing burnup of fuel in a reactor; specifically, the number of megawatt-days heat output per metric ton of uranium in the reactor fuel.

NAS:
U.S. National Academy of Sciences.

NASA:
U.S. National Aeronautics and Space Administration.

NCRP:
National Council on Radiation Protection and Measurement.

NEPA:
National Environmental Policy Act of 1969.

neutron flux:
a measure of the number of neutrons crossing a unit space in a unit of time, usually expressed in units of neutrons per square centimeter per second.

NNWSI:
Nevada Nuclear Waste Storage Investigations.

NRC:
U.S. Nuclear Regulatory Commission. Also, National Research Council of the National Academy of Sciences.

NTS:
Nevada Test Site.

NWTS Program:
Nuclear Waste Terminal Storage Program. DOE's program for the disposal of high-level nuclear waste.

ONWI:
Office of Nuclear Waste Isolation at Battelle Memorial Institute, Columbus, Ohio, under contract to DOE.

ORNL:
Oak Ridge National Laboratory, Oak Ridge, Tennessee.

overpack:
secondary (or additional) external containment for packaged nuclear waste See also: multibarrier system.

pathway:
routes by which wastes might reach the accessible environment.

PHWR:
pressurized heavy water reactor.

plutonic rock:
a rock formed at considerable depth by crystallization of molten rock or by chemical alteration.

PNL:
Pacific Northwest Laboratory, Battelle.

poison:
any material of high neutron absorption cross-section that absorbs neutrons prior to their interaction in a fission reaction.

PWR:
pressurized water reactor. A reactor system that uses a pressurized water primary cooling system.

rad:
radiation absorbed dose. The basic unit of absorbed dose of ionizing radiation. A dose of one rad means the absorption of 100 ergs of radiation energy per gram of absorbing material.

radioactive particulates:
minute discrete particles which are radioactive.

radiolysis:
splitting of chemical molecules due to interactions with radiation.

radionuclide:
the radioactive form of an element which exhibits spontaneous decay or disintegration, usually accompanied by the emission of ionizing radiation.

radiotoxicity:
a measure of the ability to cause health effects due to radiation.

R&D:
research and development.

release limits:
regulatory limits established for nuclear facilities regarding permissible amounts of radioactive material released into the atmosphere or water.

rem:
the unit of individual dose equivalent.

repository system:
the configuration of man-made features designed to act in harmony with the natural system to provide long-term containment and isolation of nuclear wastes and to provide for receipt, inspection handling, emplacement, and potential retrieval of wastes during the operating phase.

reracking:
a rearrangement of the water pool used for storage of spent fuel which results in additional spent-fuel storage capacity.

residual uncertainties:
those inherent uncertainties in data, modeling, and assumed future conditions which cannot be eliminated.

retrievability:
capability to remove waste from its place of isolation using planned engineering procedures.

risk:
the chance of consequence, often expressed as the product of the consequences and the probability of an event's occurrence.

roentgen:
a unit for measuring gamma or x-ray radiation. The roentgen is defined by measuring the effect of the radiation on air. It is that amount of gamma or x-rays required to produce ions carrying 1 electrostatic unit of charge in 1 cubic centimeter of dry air at standard conditions of temperature and pressure.

salt dome:
natural intrusive diapiritic salt formations.

seismicity:
the tendency for the occurrance of earthquakes.

sensitivity analysis:
a method of determining the importance of changes in parameters or assumptions relative to a predicted outcome.

shielding:
materials used to absorb radiation. Common shielding materials are concrete, water, and lead.

shipping cask:
a specially designed container used for shipping radioactive materials.

sipping method:
procedure by which the amount and character of gases escaping from a defective fuel assembly are measured.

sorption:
retardation of chemicals in solution by adsorption or absorption; the term for retention of one substance by another, by close-range chemical or physical forces. Absorption occurs within the pores of a material; adsorption occurs chiefly at the surface of a material.

source term:
a definition used in mathematical modeling to describe input into a system.

spall:
to flake off from a surface.

spent fuel:
fuel utilized to the extent that it can no longer be efficiently used in a nuclear reactor.

storage rack:
the rack in the water storage pool that holds the unencapsulated spent fuel assemblies.

subcritical:
the state wherein fissionable material is insufficient in quantity or of improper geometry to sustain a . . . chain reaction.

synergistic:
pertaining to the actions of more than one agent, whose combined effect is greater than and/or different from the sum of the individual effects.

tectonic process:
any process that produces changes in the Earth's crust as a result of the application of stress.

thermal power level:
a measure of the heat given off by a material to its environment.

transmutation:
conversion of a isotope to a different isotope by bombarding it with nuclear particles.

transshipment:
shipping spent fuel from one reactor of a utility company to the site of another reactor of the same type owned by the same company for the purpose of storage at the second reactor site.

transuranic:
TRU; pertaining to elements with mass numbers greater than 92, including (among others) neptunium, plutonium, americium, and curium.

TRU waste:
transuranic waste. DOE defines TRU waste as that waste containing amounts of transuranic elements greater than 10 nanocuries per gram.

tuff:
a rock composed of the finer kinds of volcanic detritus, usually stratified.

ultrasonic:
waves or vibrations having a frequency of about 20,000 hertz or more.

USDA:
U.S. Department of Agriculture.

waste management:
the planning, execution, and surveillance of essential functions related to the control of radioactive (and nonradioactive) waste, including treatment, solidification, initial or long-term storage, surveillance, and isolation.

waste package:
the system of engineered components that may include waste form, stabilizer, canister, overpack, sleeve, and emplacement hole backfill.

water basin storage:
a specially designed and operated large water pool for storing, cooling, maintaining, and shielding spent fuel elements.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF PROPOSED RULEMAKING ON THE STORAGE AND DISPOSAL OF NUCLEAR WASTE (Waste Confidence Rulemaking)	)	PR-50, 51 (44 FR 61372)

CERTIFICATE OF SERVICE

I hereby certify that I have served two copies (unless otherwise noted) of the foregoing document, entitled "Statement of Position of the United States Department of Energy," by mail or hand delivery, upon the following, this 15th day of April 1980:

Marshall E. Miller, Esq.
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555
(one copy)

Docketing and Service Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(Original and 39 copies by hand)

E. Leo Slaggie, Esq.
Office of the General Counsel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(one copy)

Karen D. Cyr, Esq.
Rulemaking and Enforcement Division
Office of the Executive Legal Director
MNBB 9604
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(one copy)

David Santee Miller, Esq.
213 Morgan Street, N.W.
Washington, D.C. 20001

Mr. Eugene N. Cramer
Neighbors of the Environment
17146 Ridgepark
Hacienda Heights, California
91745

Richard M. Sandvik, Esq.
Assistant Attorney General
500 Pacific Building
520 S. W. Yamhill
Portland, Oregon 97204

Karin P. Sheldon, Esq.
Sheldon, Harmon and Weiss
Suite 506
1725 I Street, N.W.
Washington, D.C. 20006

Mr. James C. Malaro, Chief
High Level Waste Licensing Management Branch
Office of Nuclear Material Safety and Safeguards, Mail Stop 905-SS
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(one copy)

Mr. Edward P. Regnier
Mail Stop 906-SS
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(one copy)

Keith A. Onsdorff, Esq.
Assistant Deputy Public Advocate
Division of Public Interest Advocacy
P. O. Box 141
Trenton, New Jersey 08635

Mrs. W. W. Schaefer
Safe Haven, Ltd.
3741 Koehler Drive
Sheboygan, Wisconsin 53081

Maurice Axelrad, Esq.
Lowenstein, Newman, Reis, Axelrad and Toll
1025 Connecticut Avenue, N.W.
Washington, D.C. 20036

E. Dennis Muchnicki, Esq.
Assistant Attorney General
Environmental Law Section
30 East Broad Street, 17th Floor
Columbus, Ohio 43215

Jocelyn F. Olson, Esq.
Special Assistant Attorney General
1935 West County Road B2
Roseville, Minnesota 55113

E. Tupper Kinder, Esq.
Assistant Attorney General
Environmental Protection Division
State House Annex
25 Capitol Street
Concord, New Hampshire 03301

Mr. Marvin L. Lewis
6504 Bradford Terrace
Philadelphia, Pennsylvania 19149

Dr. Judith Johnsrud
Environmental Coalition on Nuclear Power
433 Orlando Avenue
State College, Pennsylvania 16801

Ronald J. Wilson, Esq.
810 18th St., N.W.
Washington, D.C. 20006

Ezra I. Bialik, Esq.
Assistant Attorney General
Environmental Protection Bureau
Two World Trade Center
New York, New York 10047

Mr. Michael H. Raudenbush
The S.M. Stoller Corporation
1919 14th Street, Suite 500
Boulder, Colorado 80302
(one copy)

Dr. William A. Lochstet
119 E. Aaron Drive
State College, Pennsylvania 16801

Richard P. Wilson, Esq.
Assistant Attorney General
2600 Bull Street
Columbia, South Carolina 29201

Elliott Andalman, Esq.
Andalman, Adelman & Steiner, P.A.
224 Second Avenue
Hattiesburg, Mississippi 39401

Harvey Price, Esq.
General Counsel
Atomic Industrial Forum, Inc.
7101 Wisconsin Avenue
Washington, D.C. 20014

Mr. Creg Darby
Hanford Conversion Project
1817 N.E. 17th
Portland, Oregon 97212

Mr. A. S. West
Rohm and Haas Company
Independence Mall West
Philadelphia, Pennsylvania 19105

Raymond M. Momboisse, Esq.
Pacific Legal Foundation
1990 M Street, N.W.
Washington, D.C. 20036

Mr. James R. Richards
Capital Legal Foundation
1101 17th Street, N.W., Suite 810
Washington, D.C. 20036

Mr. Orville Hill
2315 Camas Avenue
Richland, Washington 99352

Mr. David Berick
Environmental Policy Institute
317 Pennsylvania Avenue, S.E.
Washington, D.C. 20003

Christopher Ellison, Esq.
California Energy Commission
1111 Howe Avenue
Sacramento, California 95825

Dr. Bertram Wolfe
Vice President and General Manager
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Mr. Ken Kramer
Lone Star Chapter of the
Sierra Club
P.O. Box 1931
Austin, Texas 78767

Ms. Priscilla C. Grew
Director, Department of
Conservation
State of California
1416 Ninth Street
Sacramento, California 95814

Carl Valore, Jr., Esq.
Valore, McAllister, Aron and
Westermoreland
Mainland Professional Plaza
535 Tilton Road
Northfield, New Jersey 08225

Richard W. Lowerre, Esq.
Assistant Attorney General
Environmental Protection Division
P.O. Box 12548, Capitol
Station
Austin, Texas 78711

James P. McGranery, Jr., Esq.
LeBoeuf, Lamb, Leiby and
MacRae
1333 New Hampshire Ave., N.W.
Washington, D.C. 20036

Dr. Miro M. Todorovich
Executive Secretary
Scientists and Engineers for
Secure Energy
410 Riverside Drive, Suite 82A
New York, New York 10025

George C. Freeman, Jr., Esq.
Hunton & Williams
P.O. Box 1535
707 Main Street
Richmond, Virginia 23212

Mr. Robert Halstead
Department of Administration
State of Wisconsin
1 West Wilson Street
Madison, Wisconsin 53702

R. Leonard Vance, Esq.
Assistant Attorney general
Supreme Court Building
1101 East Broad Street
Richmond, Virginia 23219

Joseph Gallo, Esq.
Isham, Lincoln and Beale
1050 17th Street, N.W., Suite 701
Washington, D.C. 20036

Michael I. Miller, Esq.
Isham, Lincoln and Beale
One First National Plaza, Suite 4200
Chicago, Illinois 60603

June D. MacArtor, Esq.
Deputy Attorney General
Tatnall Building
P.O. Box 1401
Dover, Delaware 19901

Mr. William J. Cahill, Jr.
Vice President
Consolidated Edison Company of
New York, Inc.
4 Irving Place
New York, New York 10003

Mr. Ray K. Robinson
Exxon Nuclear Company, Inc.
777 106th Avenue, N.E., C-00777
Bellevue, Washington 98009

William Griffin, Esq.
Assistant Attorney General
Office of the Attorney General
109 State Street
Montpelier, Vermont 05602

Michael J. Scibinico, II, Esq.
Assistant Attorney General
Department of Natural Resources
Tawes State Office Building
Annapolis, Maryland 21401

Richard M. Hluchan, Esq.
Deputy Attorney General
36 West State Street
Trenton, New Jersey 08625

Harry H. Voigt, Esq.
LeBoeuf, Lamb, Leiby and
MacRae
1333 New Hampshire Ave., N.W.
Washington, D.C. 20036

Ms. Lorna Salzman
Friends of the Earth
72 Jane Street
New York, New York 10014

Alvin H. Gutterman, Esq.
Tennessee Valley Authority
Knoxville, Tennessee 37902

Mr. Bryan L. Baker
Mockingbird Alliance
900 Lovett Boulevard, Suite 207
Houston, Texas 77006

Francis S. Wright, Esq.
Assistant Attorney General
Environmental Protection Division
One Ashburton Place, 19th Floor
Boston, Massachusetts 02108

Charles S. Rogers, Esq.
Assistant Attorney General
112 State Capitol
Oklahoma City, Oklahoma 73105

Robert M. Lindholm, Esq.
Assistant Attorney General
Jefferson City, Missouri 65102

Patrick Walsh, Esq.
Assistant Attorney General
Wisconsin Department of Justice
114 East, State Capitol
Madison, Wisconsin 53702

Mr. John O'Neill, II
Route 2, Box 44
Maple City, Michigan 49664

Mr. Ashton J. O'Donnell
Bechtel National, Inc.
P. O. Box 3965
San Francisco, California 94119

Mr. Phillip Warburg
State of Connecticut
Suite 317
444 North Capitol Street
Washington, D.C. 20001

Mr. Wayne McDanal
Federal Energy Regulatory
Commission
North Building, Room 3408
Washington, D.C. 20426
(one copy)

Lawrence K. Lau, Esq.
Deputy Attorney General
State Capitol
Honolulu, Hawaii 96813

Joseph B. Knotts, Esq.
Debevoise & Liberman
1200 - 17th Street, N.W.
Washington, D.C. 20036

Richard Troy, Esq.
Assistant Attorney General
Environmental Protection Division
Department of Justice
234 Loyola Building, 79th Floor
New Orleans, Louisiana 70112

John Van Vranken, Esq.
Chief, Northern Region
Environmental Control Division
Suite 2315
188 West Randolph Street
Chicago, Illinois 60601

Thomas M. Lemberg, Esq.
Leva, Hawes, Symington,
Martin and Oppenheimer
815 Connecticut Ave., N.W.
Washington, D.C. 20006
(one copy)

Mr. George DeBuchananne
Chief, Office of Radiohydrology
Geological Survey
U.S. Department of the Interior
Reston, Virginia 22092

Mr. John J. Kearney
Senior Vice President
Edison Electric Institute
1111 - 19th Street, N.W.
Washington, D.C. 20036

Lawrence Coe Lanpher, Esq.
Hill, Christoper & Phillips, P.C.
1900 M Street, N.W.
Washington, D.C. 20036

____
Omer F. Brown, II
Office of the General Counsel
Forrestal Mail Stop 6A-152
U.S. Department of Energy
1000 Independence Ave., S.W.
Washington, D.C. 20585

Note Concerning Service List

Other participants are requested to also serve documents in this proceeding upon the following persons:

Omer F. Brown, II, Esq.
Office of the General Counsel
Forrestal Mail Stop 6A-152
U.S. Department of Energy
1000 Independence Ave., S.W.
Washington, D.C. 20585

Mr. Ralph Stein
Office of Nuclear Waste Management
Mail Stop B107
U.S. Department of Energy
Washington, D.C. 20545

Mr. M. A. Glora
Site Qualification & Licensing
Department
Office of Nuclear Waste
505 King Avenue
Columbus, Ohio 43201